PSE&G Public Service Electric and Gas Company

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Robert L. Mittl General Manager
Nuclear Assurance and Regulation

August 30, 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.

Per D. Wagner

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In addition, enclosed for your review and approval (see Attachment 4) are the resolutions to the Draft SER open items listed in Attachment 3 and revised FSAR Sections 1.8.1.26, 6.2.5.2.5, 7.6.1.4.3, and Figure 8.3-16. A signed original of the required affidavit is provided to document the submittal of these items.

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

Very truly yours,

R J McDott

Attachments/Enclosure

C D. H. Wagner
USNRC Licensing Project Manager

W. H. Bateman
USNRC Senior Resident Inspector

FM05 1/2

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION
DOCKET NO. 50-354

PUBLIC SERVICE ELECTRIC AND GAS COMPANY

Public Service Electric and Gas Company hereby submits the enclosed Hope Creek Generating Station Draft Safety Evaluation Report open item responses and revised FSAR Sections 1.8.1.26, 6.2.5.2.5, 7.6.1.4.3, and Figure 8.3-16.

The matters set forth in this submittal are true to the best of my knowledge, information, and belief.

Respectfully submitted,

Public Service Electric
and Gas Company

By: Thomas J. Martin
Thomas J. Martin
Vice President -
Engineering and Construction

Sworn to and subscribed before me, a Notary Public of New Jersey, this 30th day of August 1984.

David K. Burd

DAVID K. BURD
NOTARY PUBLIC OF NEW JERSEY
My Comm. Expires 10-23-85

GJ02/5

DATE: 8/30/84

ATTACHMENT 1

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
1	2.3.1	Design-basis temperatures for safety-related auxiliary systems	Complete	8/15/84
2a	2.3.3	Accuracies of meteorological measurements	Complete	8/15/84 (Rev. 1)
2b	2.3.3	Accuracies of meteorological measurements	Complete	8/15/84 (Rev. 1)
2c	2.3.3	Accuracies of meteorological measurements	Complete	8/15/84 (Rev. 2)
2d	2.3.3	Accuracies of meteorological measurements	Complete	8/15/84 (Rev. 2)
3a	2.3.3	Upgrading of onsite meteorological measurements program (III.A.2)	Complete	8/15/84 (Rev. 2)
3b	2.3.3	Upgrading of onsite meteorological measurements program (III.A.2)	Complete	8/15/84 (Rev. 2)
3c	2.3.3	Upgrading of onsite meteorological measurements program (III.A.2)	NRC Action	
4	2.4.2.2	Ponding levels	Complete	8/03/84
5a	2.4.5	Wave impact and runup on service Water Intake Structure	Complete	8/20/84 (Rev. 1)
5b	2.4.5	Wave impact and runup on service water intake structure	Complete	8/20/84 (Rev. 1)
5c	2.4.5	Wave impact and runup on service water intake structure	Complete	7/27/84
5d	2.4.5	Wave impact and runup on service water intake structure	Complete	8/20/84 (Rev. 1)
6a	2.4.10	Stability of erosion protection structures	Complete	8/20/84
6b	2.4.10	Stability of erosion protection structures	Complete	8/20/84
6c	2.4.10	Stability of erosion protection structures	Complete	8/03/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
7a	2.4.11.2	Thermal aspects of ultimate heat sink	Complete	8/3/84
7b	2.4.11.2	Thermal aspects of ultimate heat sink	Complete	8/3/84
8	2.5.2.2	Choice of maximum earthquake for New England - Piedmont Tectonic Province	Complete	8/15/84
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

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
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
25	2.5.4	Intake structure soil modeling	Complete	8/10/84
26	2.5.4.4	Intake structure sliding stability	Complete	8/20/84
27	2.5.5	Slope stability	Complete	6/1/84
28a	3.4.1	Flood protection	Complete	8/30/84 (Rev. 1)
28b	3.4.1	Flood protection	Complete	8/30/84 (Rev. 1)
28c	3.4.1	Flood protection	Complete	8/30/84 (Rev. 1)
28d	3.4.1	Flood protection	Complete	8/30/84 (Rev. 1)
28e	3.4.1	Flood protection	Complete	8/30/84 (Rev. 1)
28f	3.4.1	Flood protection	Complete	7/27/84
28g	3.4.1	Flood protection	Complete	7/27/84
29	3.5.1.1	Internally generated missiles (outside containment)	Complete	8/3/84 (Rev. 1)
30	3.5.1.2	Internally generated missiles (inside containment)	Closed (5/30/84-Aux.Sys.Mtg.)	6/1/84
31	3.5.1.3	Turbine missiles	Complete	7/18/84
32	3.5.1.4	Missiles generated by natural phenomena	Complete	7/27/84
33	3.5.2	Structures, systems, and components to be protected from externally generated missiles	Complete	7/27/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
34	3.6.2	Unrestrained whipping pipe inside containment	Complete	7/18/84
35	3.6.2	ISI program for pipe welds in break exclusion zone	Complete	6/29/84
36	3.6.2	Postulated pipe ruptures	Complete	6/29/84
37	3.6.2	Feedwater isolation check valve operability	Complete	8/20/84
38	3.6.2	Design of pipe rupture restraints	Complete	8/20/84
39	3.7.2.3	SSI analysis results using finite element method and elastic half-space approach for containment structure	Complete	8/3/84
40	3.7.2.3	SSI analysis results using finite element method and elastic half-space approach for intake structure	Complete	8/3/84
41	3.8.2	Steel containment buckling analysis	Complete	6/1/84
42	3.8.2	Steel containment ultimate capacity analysis	Complete	8/20/84 (Rev. 1)
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	8/20/84 (Rev. 1)
46	3.8.5	ACI 349 deviations for foundations	Complete	8/20/84 (Rev. 1)
47	3.8.6	Base mat response spectra	Complete	8/10/84 (Rev. 1)
48	3.8.6	Rocking time histories	Complete	8/20/84 (Rev. 1)

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
49	3.8.6	Gross concrete section	Complete	8/20/84 (Rev. 1)
50	3.8.6	Vertical floor flexibility response spectra	Complete	8/20/84 (Rev. 1)
51	3.8.6	Comparison of Bechtel independent verification results with the design-basis results	Complete	8/20/84 (Rev. 2)
52	3.8.6	Ductility ratios due to pipe break	Complete	8/3/84
53	3.8.6	Design of seismic Category I tanks	Complete	8/20/84 (Rev. 1)
54	3.8.6	Combination of vertical responses	Complete	8/10/84 (Rev. 1)
55	3.8.6	Torsional stiffness calculation	Complete	6/1/84
56	3.8.6	Drywell stick model development	Complete	8/20/84 (Rev. 1)
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	8/20/84 (Rev. 1)
60	3.8.6	BSAP element size limitations	Complete	8/20/84 (Rev. 1)
61	3.8.6	Seismic modeling of drywell shield wall	Complete	6/1/84
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

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
64	3.8.6	SSI analysis 12 Hz cutoff frequency	Complete	8/20/84 (Rev. 1)
65	3.8.6	Intake structure crane heavy load drop	Complete	6/1/84
66	3.8.6	Impedance analysis for the intake structure	Complete	8/10/84 (Rev. 1)
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	8/20/84 (Rev. 1)
78	3.8.6	Dead load calculations	Complete	6/1/84
79	3.8.6	Post-modification seismic loads for the torus	Complete	8/20/84 (Rev. 1)

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
80	3.8.6	Torus fluid-structure interactions	Complete	6/1/84
81	3.8.6	Seismic displacement of torus	Complete	8/20/84 (Rev. 1)
82	3.8.6	Review of seismic Category I tank design	Complete	8/20/84 (Rev. 1)
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	8/20/84 (Rev. 1)
85	3.8.6	Load combination consistency	Complete	6/1/84
86	3.9.1	Computer code validation	Complete	8/20/84
87	3.9.1	Information on transients	Complete	8/20/84
88	3.9.1	Stress analysis and elastic-plastic analysis	Complete	6/29/84
89	3.9.2.1	Vibration levels for NSSS piping systems	Complete	6/29/84
90	3.9.2.1	Vibration monitoring program during testing	Complete	7/18/84
91	3.9.2.2	Piping supports and anchors	Complete	6/29/84
92	3.9.2.2	Triple flued-head containment penetrations	Complete	6/15/84
93	3.9.3.1	Load combinations and allowable stress limits	Complete	6/29/84
94	3.9.3.2	Design of SRVs and SRV discharge piping	Complete	6/29/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
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	8/20/84 (Rev. 1)
97	3.9.3.3	Buckling criteria used for component supports	Complete	6/29/84
98	3.9.3.3	Design of bolts	Complete	6/15/84
99a	3.9.5	Stress categories and limits for core support structures	Complete	6/15/84
99b	3.9.5	Stress categories and limits for core support structures	Complete	6/15/84
100a	3.9.6	10CFR50.55a paragraph (g)	Complete	6/29/84
100b	3.9.6	10CFR50.55a paragraph (g)	Complete	8/20/84
101	3.9.6	PSI and ISI programs for pumps and valves	Complete	8/20/84
102	3.9.6	Leak testing of pressure isolation valves	Complete	6/29/84
103a1	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103a2	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103a3	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103a4	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
103a5	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103a6	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103a7	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103b1	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103b2	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103b3	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103b4	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103b5	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103b6	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103c1	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103c2	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103c3	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103c4	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
104	3.11	Environmental qualification of mechanical and electrical equipment	NRC Action	

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
105	4.2	Plant-specific mechanical fracturing analysis	Complete	8/20/84 (Rev. 1)
106	4.2	Applicability of seismic andd LOCA loading evaluation	Complete	8/20/84 (Rev. 1)
107	4.2	Minimal post-irradiation fuel surveillance program	Complete	6/29/84
108	4.2	Gadolina thermal conductivity equation	Complete	6/29/84
109a	4.4.7	TMI-2 Item II.F.2	Complete	8/20/84
109b	4.4.7	TMI-2 Item II.F.2	Complete	8/20/84
110a	4.6	Functional design of reactivity control systems	Complete	8/30/84 (Rev. 1)
110b	4.6	Functional design of reactivity control systems	Complete	8/30/84 (Rev. 1)
111a	5.2.4.3	Preservice inspection program (components within reactor pressure boundary)	Complete	6/29/84
111b	5.2.4.3	Preservice inspection program (components within reactor pressure boundary)	Complete	6/29/84
111c	5.2.4.3	Preservice inspection program (components within reactor pressure boundary)	Complete	6/29/84
112a	5.2.5	Reactor coolant pressure boundary leakage detection	Complete	8/30/84 (Rev. 1)
112b	5.2.5	Reactor coolant pressure boundary leakage detection	Complete	8/30/84 (Rev. 1)

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
112c	5.2.5	Reactor coolant pressure boundary leakage detection	Complete	8/30/84 (Rev. 1)
112d	5.2.5	Reactor coolant pressure boundary leakage detection	Complete	8/30/84 (Rev. 1)
112e	5.2.5	Reactor coolant pressure boundary leakage detection	Complete	8/30/84 (Rev. 1)
113	5.3.4	GE procedure applicability	Complete	7/18/84
114	5.3.4	Compliance with NB 2360 of the Summer 1972 Addenda to the 1971 ASME Code	Complete	7/18/84
115	5.3.4	Drop weight and Charpy v-notch tests for closure flange materials	Complete	7/18/84
116	5.3.4	Charpy v-notch test data for base materials as used in shell course No. 1	Complete	7/18/84
117	5.3.4	Compliance with NB 2332 of Winter 1972 Addenda of the ASME Code	Complete	8/20/84
118	5.3.4	Lead factors and neutron fluence for surveillance capsules	Complete	8/20/84
119	6.2	TMI item II.E.4.1	Complete	6/29/84
120a	6.2	TMI Item II.E.4.2	Complete	8/20/84
120b	6.2	TMI Item II.E.4.2	Complete	8/20/84
121	6.2.1.3.3	Use of NUREG-0588	Complete	7/27/84
122	6.2.1.3.3	Temperature profile	Complete	7/27/84
123	6.2.1.4	Butterfly valve operation (post accident)	Complete	6/29/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
124a	6.2.1.5.1	RPV shield annulus analysis	Complete	8/20/84 (Rev. 1)
124b	6.2.1.5.1	RPV shield annulus analysis	Complete	8/20/84 (Rev. 1)
124c	6.2.1.5.1	RPV shield annulus analysis	Complete	8/20/84 (Rev. 1)
125	6.2.1.5.2	Design drywell head differential pressure	Complete	6/15/84
126a	6.2.1.6	Redundant position indicators for vacuum breakers (and control room alarms)	Complete	8/20/84
126b	6.2.1.6	Redundant position indicators for vacuum breakers (and control room alarms)	Complete	8/20/84
127	6.2.1.6	Operability testing of vacuum breakers	Complete	8/20/84 (Rev. 1)
128	6.2.2	Air ingestion	Complete	7/27/84
129	6.2.2	Insulation ingestion	Complete	6/1/84
130	6.2.3	Potential bypass leakage paths	Complete	6/29/84
131	6.2.3	Administration of secondary containment openings	Complete	7/18/84
132	6.2.4	Containment isolation review	Complete	6/15/84
133a	6.2.4.1	Containment purge system	Complete	8/20/84
133b	6.2.4.1	Containment purge system	Complete	8/20/84
133c	6.2.4.1	Containment purge system	Complete	8/20/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
134	6.2.6	Containment leakage testing	Complete	6/15/84
135	6.3.3	LPCS and LPCI injection valve interlocks	Complete	8/20/84
136	6.3.5	Plant-specific LOCA (see Section 15.9.13)	Complete	8/20/84 (Rev. 1)
137a	6.4	Control room habitability	Complete	8/20/84
137b	6.4	Control room habitability	Complete	8/20/84
137c	6.4	Control room habitability	Complete	8/20/84
138	6.6	Preservice inspection program for Class 2 and 3 components	Complete	6/29/84
139	6.7	MSIV leakage control system	Complete	6/29/84
140a	9.1.2	Spent fuel pool storage	Complete	8/15/84 (Rev. 1)
140b	9.1.2	Spent fuel pool storage	Complete	8/15/84 (Rev. 1)
140c	9.1.2	Spent fuel pool storage	Complete	8/15/84 (Rev. 1)
140d	9.1.2	Spent fuel pool storage	Complete	8/15/84 (Rev. 1)
141a	9.1.3	Spent fuel cooling and cleanup system	Complete	8/30/84 (Rev. 1)
141b	9.1.3	Spent fuel cooling and cleanup system	Complete	8/30/84 (Rev. 1)
141c	9.1.3	Spent fuel pool cooling and cleanup system	Complete	8/30/84 (Rev. 1)

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
141d	9.1.3	Spent fuel pool cooling and cleanup system	Complete	8/30/84 (Rev. 1)
141e	9.1.3	Spent fuel pool cooling and cleanup system	Complete	8/30/84 (Rev. 1)
141f	9.1.3	Spent fuel pool cooling and cleanup system	Complete	8/30/84 (Rev. 1)
141g	9.1.3	Spent fuel pool cooling and cleanup system	Complete	8/30/84 (Rev. 1)
142a	9.1.4	Light load handling system (related to refueling)	Complete	8/15/84 (Rev. 1)
142b	9.1.4	Light load handling system (related to refueling)	Complete	8/15/84 (Rev. 1)
143a	9.1.5	Overhead heavy load handling	Open	
143b	9.1.5	Overhead heavy load handling	Open	
144a	9.2.1	Station service water system	Complete	8/15/84 (Rev. 1)
144b	9.2.1	Station service water system	Complete	8/15/84 (Rev. 1)
144c	9.2.1	Station service water system	Complete	8/15/84 (Rev. 1)
145	9.2.2	ISI program and functional testing of safety and turbine auxiliaries cooling systems	Closed (5/30/84-Aux.Sys.Mtg.)	6/15/84
146	9.2.6	Switches and wiring associated with HPCI/RCIC torus suction	Closed (5/30/84-Aux.Sys.Mtg.)	6/15/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
147a	9.3.1	Compressed air systems	Complete	8/3/84 (Rev 1)
147b	9.3.1	Compressed air systems	Complete	8/3/84 (Rev 1)
147c	9.3.1	Compressed air systems	Complete	8/3/84 (Rev 1)
147d	9.3.1	Compressed air systems	Complete	8/3/84 (Rev 1)
148	9.3.2	Post-accident sampling system (II.B.3)	Complete	8/20/84
149a	9.3.3	Equipment and floor drainage system	Complete	7/27/84
149b	9.3.3	Equipment and floor drainage system	Complete	7/27/84
150	9.3.6	Primary containment instrument gas system	Complete	8/3/84 (Rev. 1)
151a	9.4.1	Control structure ventilation system	Complete	8/30/84 (Rev. 1)
151b	9.4.1	Control structure ventilation system	Complete	8/30/84 (Rev. 1)
152	9.4.4	Radioactivity monitoring elements	Closed (5/30/84- Aux.Sys.Mtg.)	6/1/84
153	9.4.5	Engineered safety features ventilation system	Complete	8/30/84 (Rev 2)
154	9.5.1.4.a	Metal roof deck construction classificiation	Complete	6/1/84
155	9.5.1.4.b	Ongoing review of safe shutdown capability	NRC Action	
156	9.5.1.4.c	Ongoing review of alternate shutdown capability	NRC Action	

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
157	9.5.1.4.e	Cable tray protection	Complete	8/20/84
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
160	9.5.1.5.b	Fire water pump capacity	Complete	8/13/84
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 protection	Complete	6/1/84
166	12.3.4.2	Airborne radioactivity monitor positioning	Complete	7/18/84
167	12.3.4.2	Portable continuous air monitors	Complete	7/18/84
168	12.5.2	Equipment, training, and procedures for inplant iodine instrumentation	Complete	6/29/84
169	12.5.3	Guidance of Division B Regulatory Guides	Complete	7/18/84
170	13.5.2	Procedures generation package submittal	Complete	6/29/84
171	13.5.2	TMI Item I.C.1	Complete	6/29/84
172	13.5.2	PGP Commitment	Complete	6/29/84
173	13.5.2	Procedures covering abnormal releases of radioactivity	Complete	6/29/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
174	13.5.2	Resolution explanation in FSAR of TMI Items I.C.7 and I.C.8	Complete	6/15/84
175	13.6	Physical security	Open	
176a	14.2	Initial plant test program	Complete	8/13/84
176b	14.2	Initial plant test program	Complete	8/13/84
176c	14.2	Initial plant test program	Complete	7/27/84
176d	14.2	Initial plant test program	Complete	8/24/84 (Rev. 2)
176e	14.2	Initial plant test program	Complete	7/27/84
176f	14.2	Initial plant test program	Complete	8/13/84
176g	14.2	Initial plant test program	Complete	8/20/84
176h	14.2	Initial plant test program	Complete	8/13/84
176i	14.2	Initial plant test program	Complete	7/27/84
177	15.1.1	Partial feedwater heating	Complete	8/20/84 (Rev. 1)
178	15.6.5	LOCA resulting from spectrum of postulated piping breaks within RCP	NRC Action	
179	15.7.4	Radiological consequences of fuel handling accidents	NRC Action	
180	15.7.5	Spent fuel cask drop accidents	NRC Action	
181	15.9.5	TMI-2 Item II.K.3.3	Complete	6/29/84
182	15.9.10	TMI-2 Item II.K.3.18	Complete	6/1/84
183	18	Hope Creek DCRDR	Complete	8/15/84

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ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
184	7.2.2.1.e	Failures in reactor vessel level sensing lines	Complete	8/1/84 (Rev 1)
185	7.2.2.2	Trip system sensors and cabling in turbine building	Complete	6/1/84
186	7.2.2.3	Testability of plant protection systems at power	Complete	8/13/84 (Rev. 1)
187	7.2.2.4	Lifting of leads to perform surveillance testing	Complete	8/3/84
188	7.2.2.5	Setpoint methodology	Complete	8/1/84
189	7.2.2.6	Isolation devices	Complete	8/1/84
190	7.2.2.7	Regulatory Guide 1.75	Complete	6/1/84
191	7.2.2.8	Scram discharge volume	Complete	6/29/84
192	7.2.2.9	Reactor mode switch	Complete	8/15/84 (Rev. 1)
193	7.3.2.1.10	Manual initiation of safety systems	Complete	8/1/84
194	7.3.2.2	Standard review plan deviations	Complete	8/1/84 (Rev 1)
195a	7.3.2.3	Freeze-protection/water filled instrument and sampling lines and cabinet temperature control	Complete	8/1/84
195b	7.3.2.3	Freeze-protection/water filled instrument and sampling lines and cabinet temperature control	Complete	8/1/84
196	7.3.2.4	Sharing of common instrument taps	Complete	8/1/84
197	7.3.2.5	Microprocessor, multiplexer and computer systems	Complete	8/1/84 (Rev 1)

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
198	7.3.2.6	TMI Item II.K.3.18-ADS actuation	Complete	8/20/84
199	7.4.2.1	IE Bulletin 79-27-Loss of non-class IE instrumentation and control power system bus during operation	Complete	8/24/84 (Rev. 1)
200	7.4.2.2	Remote shutdown system	Complete	8/15/84 (Rev 1)
201	7.4.2.3	RCIC/HPCI interactions	Complete	8/3/84
202	7.5.2.1	Level measurement errors as a result of environmental temperature effects on level instrumentation reference leg	Complete	8/3/84
203	7.5.2.2	Regulatory Guide 1.97	Complete	8/3/84
204	7.5.2.3	TMI Item II.F.1 - Accident monitoring	Complete	8/1/84
205	7.5.2.4	Plant process computer system	Complete	6/1/84
206	7.6.2.1	High pressure/low pressure interlocks	Complete	7/27/84
207	7.7.2.1	HELBs and consequential control system failures	Complete	8/24/84 (Rev. 1)
208	7.7.2.2	Multiple control system failures	Complete	8/24/84 (Rev. 1)
209	7.7.2.3	Credit for non-safety related systems in Chapter 15 of the FSAR	Complete	8/1/84 (Rev 1)
210	7.7.2.4	Transient analysis recording system	Complete	7/27/84
211a	4.5.1	Control rod drive structural materials	Complete	7/27/84
211b	4.5.1	Control rod drive structural materials	Complete	7/27/84
211c	4.5.1	Control rod drive structural materials	Complete	7/27/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
211d	4.5.1	Control rod drive structural materials	Complete	7/27/84
211e	4.5.1	Control rod drive structural materials	Complete	7/27/84
212	4.5.2	Reactor internals materials	Complete	7/27/84
213	5.2.3	Reactor coolant pressure boundary material	Complete	7/27/84
214	6.1.1	Engineered safety features materials	Complete	7/27/84
215	10.3.6	Main steam and feedwater system materials	Complete	7/27/84
216a	5.3.1	Reactor vessel materials	Complete	7/27/84
216b	5.3.1	Reactor vessel materials	Complete	7/27/84
217	9.5.1.1	Fire protection organization	Complete	8/15/84
218	9.5.1.1	Fire hazards analysis	Complete	6/1/84
219	9.5.1.2	Fire protection administrative controls	Complete	8/15/84
220	9.5.1.3	Fire brigade and fire brigade training	Complete	8/15/84
221	8.2.2.1	Physical separation of offsite transmission lines	Complete	8/1/84
222	8.2.2.2	Design provisions for re-establishment of an offsite power source	Complete	8/1/84
223	8.2.2.3	Independence of offsite circuits between the switchyard and class IE buses	Complete	8/1/84
224	8.2.2.4	Common failure mode between onsite and offsite power circuits	Complete	8/1/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
225	8.2.3.1	Testability of automatic transfer of power from the normal to preferred power source	Complete	8/1/84
226	8.2.2.5	Grid stability	Complete	8/13/84 (Rev. 1)
227	8.2.2.6	Capacity and capability of offsite circuits	Complete	8/1/84
228	8.3.1.1(1)	Voltage drop during transient conditions	Complete	8/1/84
229	8.3.1.1(2)	Basis for using bus voltage versus actual connected load voltage in the voltage drop analysis	Complete	8/1/84
230	8.3.1.1(3)	Clarification of Table 8.3-11	Complete	8/1/84
231	8.3.1.1(4)	Undervoltage trip setpoints	Complete	8/1/84
232	8.3.1.1(5)	Load configuration used for the voltage drop analysis	Complete	8/1/84
233	8.3.3.4.1	Periodic system testing	Complete	8/1/84
234	8.3.1.3	Capacity and capability of onsite AC power supplies and use of administrative controls to prevent overloading of the diesel generators	Complete	8/1/84
235	8.3.1.5	Diesel generators load acceptance test	Complete	8/1/84
236	8.3.1.6	Compliance with position C.6 of RG 1.9	Complete	8/1/84
237	8.3.1.7	Decription of the load sequencer	Complete	8/1/84
238	8.2.2.7	Sequencing of loads on the offsite power system	Complete	8/1/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
239	8.3.1.8	Testing to verify 80% minimum voltage	Complete	8/15/84
240	8.3.1.9	Compliance with BTP-PSB-2	Complete	8/1/84
241	8.3.1.10	Load acceptance test after prolonged no load operation of the diesel generator	Complete	8/20/84 (Rev. 1)
242	8.3.2.1	Compliance with position 1 of Regulatory Guide 1.128	Complete	8/1/84
243	8.3.3.1.3	Protection or qualification of Class 1E equipment from the effects of fire suppression systems	Complete	8/1/84
244	8.3.3.3.1	Analysis and test to demonstrate adequacy of less than specified separation	Complete	8/30/84 (Rev. 1)
245	8.3.3.3.2	The use of 18 versus 36 inches of separation between raceways	Complete	8/15/84 (Rev. 1)
246	8.3.3.3.3	Specified separation of raceways by analysis and test	Complete	8/1/84
247	8.3.3.5.1	Capability of penetrations to withstand long duration short circuits at less than maximum or worst case short circuit	Complete	8/1/84
248	8.3.3.5.2	Separation of penetration primary and backup protections	Complete	8/1/84
249	8.3.3.5.3	The use of bypassed thermal overload protective devices for penetration protections	Complete	8/1/84
250	8.3.3.5.4	Testing of fuses in accordance with R.G. 1.63	Complete	8/1/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
251	8.3.3.5.5	Fault current analysis for all representative penetration circuits	Complete	8/1/84
252	8.3.3.5.6	The use of a single breaker to provide penetration protection	Complete	8/1/84
253	8.3.3.1.4	Commitment to protect all Class 1E equipment from external hazards versus only class 1E equipment in one division	Complete	8/1/84
254	8.3.3.1.5	Protection of class 1E power supplies from failure of unqualified class 1E loads	Complete	8/1/84
255	8.3.2.2	Battery capacity	Complete	8/1/84
256	8.3.2.3	Automatic trip of loads to maintain sufficient battery capacity	Complete	8/20/84
257	8.3.2.5	Justification for a 0 to 13 second load cycle	Complete	8/1/84
258	8.3.2.6	Design and qualification of DC system loads to operate between minimum and maximum voltage levels	Complete	8/1/84
259	8.3.3.3.4	Use of an inverter as an isolation device	Complete	8/1/84
260	8.3.3.3.5	Use of a single breaker tripped by a LOCA signal used as an isolation device	Complete	8/1/84
261	8.3.3.3.6	Automatic transfer of loads and interconnection between redundant divisions	Complete	8/1/84
262	11.4.2.d	Solid waste control program	Complete	8/20/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
263	11.4.2.e	Fire protection for solid radwaste storage area	Complete	8/13/84
264	6.2.5	Sources of oxygen	Complete	8/20/84
265	6.8.1.4	ESF Filter Testing	Complete	8/13/84
266	6.8.1.4	Field leak tests	Complete	8/13/84
267	6.4.1	Control room toxic chemical detectors	Complete	8/13/84
268		Air filtration unit drains	Complete	8/20/84
269	5.2.2	Code cases N-242 and N-242-1	Complete	8/20/84
270	5.2.2	Code case N-252	Complete	8/20/84
TS-1	2.4.14	Closure of watertight doors to safety-related structures	Open	
TS-2	4.4.4	Single recirculation loop operation	Open	
TS-3	4.4.5	Core flow monitoring for crud effects	Complete	6/1/84
TS-4	4.4.6	Loose parts monitoring system	Open	
TS-5	4.4.9	Natural circulation in normal operation	Open	
TS-6	6.2.3	Secondary containment negative pressure	Open	
TS-7	6.2.3	Inleakage and drawdown time in secondary containment	Open	
TS-8	6.2.4.1	Leakage integrity testing	Open	
TS-9	6.3.4.2	ECCS subsystem periodic component testing	Open	

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
TS-10	6.7	MSIV leakage rate		
TS-11	15.2.2	Availability, setpoints, and testing of turbine bypass system	Open	
TS-12	15.6.4	Primary coolant activity		
LC-1	4.2	Fuel rod internal pressure criteria	Complete	6/1/84
LC-2	4.4.4	Stability analysis submitted before second-cycle operation	Open	

ATTACHMENT 2 DATE: 8/30/84

DRAFT SEP SECTIONS AND DATES PROVIDED

SECTION	DATE	SECTION	DATE
3.1			
3.2.1		11.4.1	See Notes 1&5
3.2.2		11.4.2	See Notes 1&5
5.1		11.5.1	See Notes 1&5
5.2.1		11.5.2	See Notes 1&5
6.5.1	See Notes 1&5	13.1.1	See Note 4
8.1	See Note 2	13.1.2	See Note 4
8.2.1	See Note 2	13.2.1	See Note 4
8.2.2	See Note 2	13.2.2	See Note 4
8.2.3	See Note 2	13.3.1	See Note 4
8.2.4	See Note 2	13.3.2	See Note 4
8.3.1	See Note 2	13.3.3	See Note 4
8.3.2	See Note 2	13.3.4	See Note 4
8.4.1	See Note 2	13.4	See Note 4
8.4.2	See Note 2	13.5.1	See Note 4
8.4.3	See Note 2	15.2.3	
8.4.5	See Note 2	15.2.4	
8.4.6	See Note 2	15.2.5	
8.4.7	See Note 2	15.2.6	
8.4.8	See Note 2	15.2.7	
9.5.2	See Note 3	15.2.8	
9.5.3	See Note 3	15.7.3	See Notes 1&5
9.5.7	See Note 3	17.1	8/3/84
9.5.8	See Note 3	17.2	8/3/84
10.1	See Note 3	17.3	8/3/84
10.2	See Note 3	17.4	8/3/84
10.2.3	See Note 3		
10.3.2	See Note 3		
10.4.1	See Note 3		
10.4.2	See Notes 3&5		
10.4.3	See Notes 3&5		
10.4.4	See Note 3		
11.1.1	See Notes 1&5		
11.1.2	See Notes 1&5		
11.2.1	See Notes 1&5		
11.2.2	See Notes 1&5		
11.3.1	See Notes 1&5		
11.3.2	See Notes 1&5		

Notes:

1. Open items provided in letter dated July 24, 1984 (Schwencer to Mittl)
2. Open items provided in June 6, 1984 meeting
3. Open items provided in April 17-18, 1984 meeting
4. Open items provided in May 2. 1984 meting
5. Draft SER Section provided in letter dated August 7, 1984 (Schwencer to Mittl)

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Date: August 30, 1984

ATTACHMENT 3

Open Item	DSER Section	Subject
28A-e	3.4.1	Flood Protection
110	4.6	Functional design of reactivity control systems
112	5.2.5	Reactor coolant pressure boundary leakage detection
141	9.1.3	Spent fuel pool cooling and cleanup system
151	9.4.1	Control structure ventilation system
153	9.4.5	Engineered safety features ventilation system
244	8.3.3.3.1	Analysis and test to demonstrate adequacy of less than specified separation

ATTACHMENT 4

1.8.1.23 Conformance to Regulatory Guide 1.23 (Safety Guide 8), Revision 0, February 17, 1972: Onsite Meteorological Programs

HCGS complies with Regulatory Guide 1.23.

1.8.1.24 Conformance to Regulatory Guide 1.24 (Safety Guide 24), Revision 0, March 23, 1972: Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure

Regulatory Guide 1.24 is not applicable to HCGS.

1.8.1.25 Conformance to Regulatory Guide 1.25 (Safety Guide 25), Revision 0, March 23, 1972: Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors

HCGS complies with Regulatory Guide 1.25.

1.8.1.26 Conformance to Regulatory Guide 1.26, Revision 3, February 1976: Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants

HCGS complies with Regulatory Guide 1.26, with the clarifications outlined below.

~~PSE&G's position is that equipment that is important to safety is safety-related and therefore does not distinguish between these terms.~~ PSE&G does recognize the need for the assurance of the specified operation of certain non-safety-related structures, systems and components, such as fire protection systems, radioactive waste treatment, handling and storage systems, and Seismic Category II/I items. Such assurance is documented through the specification of limited quality assurance programs (described in Table 3.2-1, footnotes (22), (50) and (52). In addition, items designated "D+" in Table 3.2-1 will be included in the QA program during operations.

The exception to Position C.2.b is that since the reactor recirculation pumps do not perform any safety function and since failure of the reactor coolant pumps due to seal or cooling water failure does not have serious safety implications, the control rod drive (CRD) seal purge supply and reactor auxiliaries cooling isolation signal. The isolation signal to the valves can be overridden manually from the main control room. Containment isolation is discussed further in Section 6.2.4. INSERT A

Each analyzer package can only sample one sample point at one time. The selection of a specific sample point is determined by the operator. Gases from the selected sample point are routed in parallel through a hydrogen analyzer cell and oxygen analyzer cell located in the analyzer panel inside the reactor building.

The operation of the hydrogen and oxygen analyzer cells is based on the measurement of thermal conductivity of the gas sample. The thermal conductivity of the gas mixture changes proportionally to the changes in the concentration of the individual gas constituents of the mixture. The thermal conductivity of hydrogen is far greater (approximately seven times the thermal conductivity of air) than any other gas expected to be present in the primary containment. The hydrogen analyzer cell incorporates a catalytic combustion feature in which hydrogen in the sample is removed by catalytic recombination with a reagent gas (oxygen). The thermal conductivity of the sample is measured before and after recombination, and the two measurements are compared. The difference in thermal conductivity is proportional to the concentration of hydrogen originally in the sample. The oxygen analyzer operates simultaneously in a similar manner, except that the reagent gas is hydrogen.

The hydrogen analyzer has dual range capability of 0 to 10% by volume and 0 to 30% by volume. The oxygen analyzer has dual range capability of 0 to 10% by volume and 0 to 25% by volume. The hydrogen and oxygen concentrations in the sample gas are indicated at the analyzer panel in the reactor building and at the remote control panel in the main control room. The concentrations are also recorded in the main control room. An additional oxygen indication is provided at the entrance to the drywell service hatch.

Sample gases are drawn through the analyzer cells by the diaphragm pump located in the analyzer panel. Sample gases and any excess moisture, either from the sample or created by the catalytic recombination, are routed back to the suppression chamber.

HOAS design and performance data is included in Table 6.2-17. The HOAS environmental qualification program is found in

INSERT A PAGE 6.2-75

Following a postulated LOCA, one HOAS channel is manually initiated and operates continuously for the duration of the accident.

four LPRM strings (16 detectors) surrounding the selected rod are used in the RBM to provide protection against local fuel overpower conditions.

7.6.1.4.3 Average Power Range Monitor Subsystem

The APRM subsystem monitors neutron flux from approximately 1% to above 100% power. There are six APRM channels, each receiving core flux level signals from 21 or 22 LPRM detectors. Each APRM channel averages the 21 or 22 separate neutron flux signals from the LPRMs assigned to it, and generates a signal representing core average power.

This signal is used to drive a local meter and a remote recorder located on the main control room vertical board. It is also applied to a trip unit to provide APRM downscale, inoperative and upscale alarms, and upscale reactor trip signals for use in the RPS or RMCS.

Refer to Section 7.2.1.1 for a description of the APRM inputs to the RPS, and Figure 7.6-5 for the RPS trip circuit input arrangement. APRM trips are summarized in Table 7.6-2.

The APRM scram units are set for a reactor scram at 15% core power in "refuel" and "startup" modes. When the mode switch is in "run," the APRM trip reference signal is provided by a signal that varies with recirculation flow. This provides a power following reactor scram setpoint. As power increases, the reactor scram setpoint also increases up to a fixed setpoint above 100%. Reactor power is always bounded with a reactor scram, yet the change in power required to generate the reactor scram does not vary greatly with the operating power level.

Provision is made for manually bypassing one APRM channel at a time. Calibration or maintenance can be performed without tripping the RPS. Removal of an APRM channel from service without bypassing it, by unplugging a card, by taking the APRM function switch out of "operate," or by having too few assigned LPRM signals to the APRM, will result in an APRM "inoperative" condition which causes a half scram, a rod block, and annunciation

The APRM channels receive power from non-Class 1E uninterruptible power sources. Power for each APRM trip unit is supplied from

(see Figure 8.3-11, sht. 3)

the same power supply as its associated APRM. The ac bus used for a given APRM channel also supplies power to its associated LPRMs.

-INSERT A-

APRM signals are sent to redundant reactivity control system (RRCS) to enable the logic if additional reactivity control is necessary following an ATWS event. The use of this signal is discussed in Section 7.6.1.7.

The APRMs are designed to remain accurately functional for at least 20 minutes after an ATWS feedwater run-back is initiated.

7.6.1.5 Recirculation Pump Trip System - Instrumentation and Controls

7.6.1.5.1 RPT Purpose

The reason for tripping the recirculation pumps is to reduce the impact on the fuel of thermal transients caused by turbine trip, generator trip, or load rejection. The rapid core flow reduction increases void content and thereby introduces negative reactivity in conjunction with control rod insertion.

7.6.1.5.2 RPT Logic and Operation

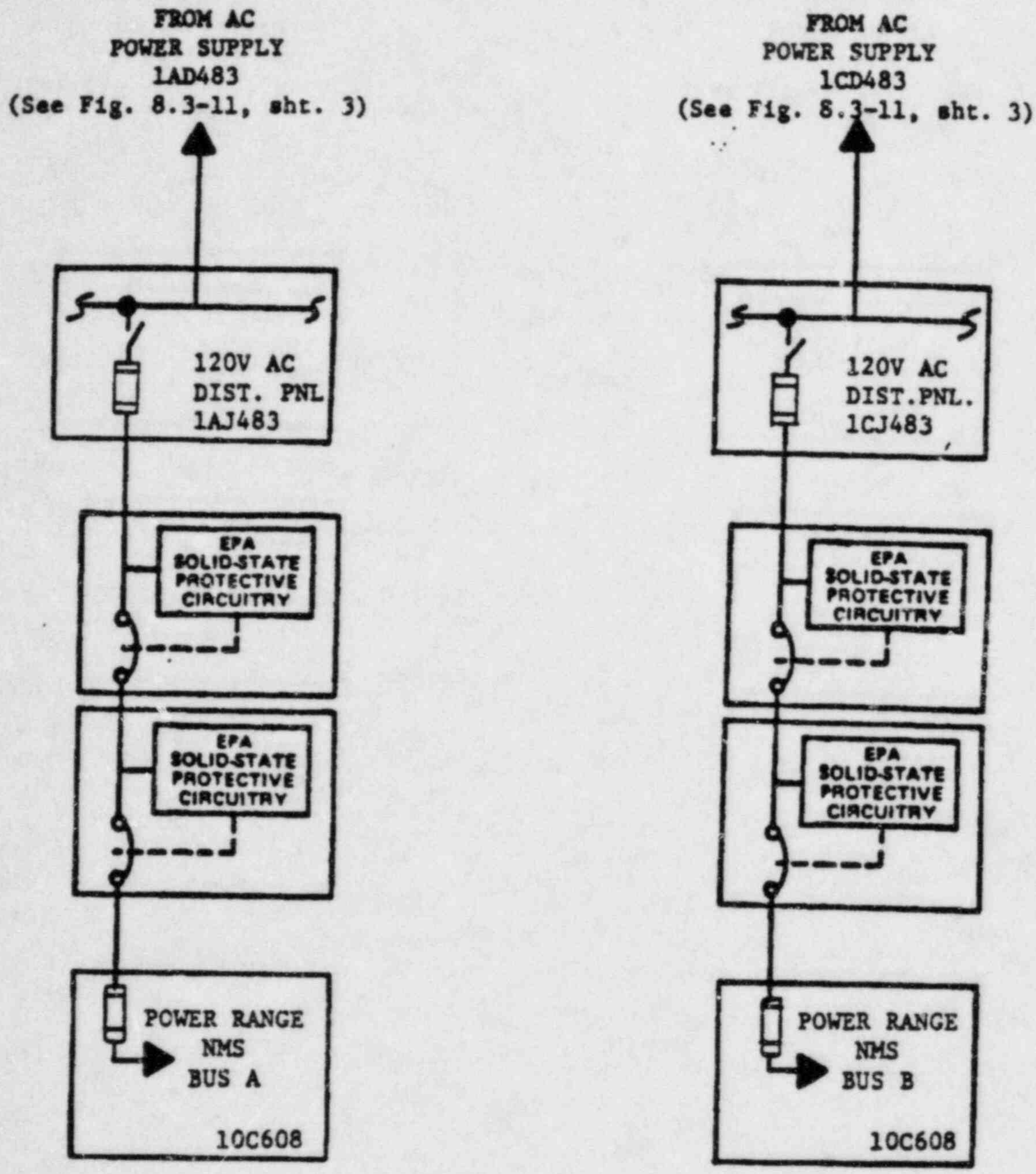
The RPS detects turbine control valve fast closure and main stop valve closure, using four channels of sensor logic. This is combined into two channelized two-out-of-two trip logic for RPT.

Trip signal initiation requires confirmation from at least two sensor channels. No single failure will prevent RPT trip.

Each trip logic channel will trip both recirculation pumps.

— INSERT A —

Electrical protection assemblies (EPAs) identical to those used in the reactor protection system (described in Section 8.3.1.5.4) are installed between the power range NMS and each of the two 120v ac feeders from the UPS power sources (see Figure 8.3-16). The EPAs ensure that the power range NMS never operates under degraded bus voltage or frequency conditions. The power range NMS panel (10C608) was analyzed with this power supply configuration to ensure that no single failure of the power range NMS could jeopardize the safe shutdown of the plant.



HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

ELECTRICAL PROTECTION ASSEMBLIES (EPAs) IN THE POWER RANGE NEUTRON MONITORING SYSTEM

FIGURE 8.3-16 Amendment

 Rev 1

a,b,c,d&e.
DSER Open Item No. 28 (DSER Section 3.4.1)

FLOOD PROTECTION

The design of the facility for flood protection was reviewed in accordance with Section 3.4.1 of the Standard Review Plan (SRP) NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the design of the facility for flood protection with respect to the applicable regulations of 10 CFR Part 50.

In order to assure conformance with the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomona," our review of the overall flood protection design included all systems and components whose failure due to flooding could prevent safe shutdown of the plant or result in uncontrolled release of significant radioactivity.

The applicant has sited the plant (at elevation 22.5 feet Mean Sea Level (MSL)) along the Delaware River near the point where the river flows into the Atlantic Ocean. The design basis flood is the result of the probable maximum hurricane (PMH) surge with wave runup coincident with the 10% exceedance high tide. The design basis flood level for all structures is 34.8 feet MSL, including wave activity (refer to Section 2.4.2 of this SER). The design basis flood level of 34.8 feet MSL represents plant submergence at the plant site by 12 feet 3.6 inches. Vertical and horizontal construction joints are provided with waterstop to elevation 32 feet MSL. [The applicant must water-proof all safety-related structures and all penetrations to those structures to a higher elevation than the flood elevation of the design basis flood (PMH).] 28a

The probable maximum flood which results in over 12.3 feet of water onsite is due to the PMH and is greater than the flooding due to the probable maximum precipitation.

The personnel access doors to areas where flood protection must be provided are all submarine doors which open outward, except doors 31B and 15B. [In order to comply with the guidelines of Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants", Position C1, the applicant must modify doors 31B and 15B to be submarine doors or equivalent for these doors to open outward or assume the doors are open during the design basis flood and verify that no safety-related equipment will be flooded.] 28b
[The applicant has not provided information requested concerning Regulatory Guide 1.102, Position C.2, and therefore no conclusions

Item No. 28 (Cont'd)

28g

can be made concerning compliances at this time.] [The applicant has not committed to providing sensors on all doors and hatches in exterior walls which are below the desgin basis flood elevation plus wind-generated wave effects to alarm in the control room when they are opened. As an alternative, the applicant may provide the results of a flooding analysis with the administratively controlled doors open and which shows that no safety-related equipment will be flooded.] - 28c

[The site contains non-seismic Category I tanks. The applicant has stated that the site drainage system will prevent the contents of the failed tanks (as the result of a safe shutdown earthquake) from flooding the safety-related structures. The applicant has not identified the site drainage system as safety-related, seismic Category I. The site drainage system must be safety-related and seismic Category I in order to take credit for the system after a design basis event. Similarly, the site drainage system should be tornado and tornado missile protected if the drainage system is needed to prevent any flooding resulting from tank(s) failure due to a tornadic even or due to tornado generated missiles.] - 28d

The applicant has stated that the electrical cables will continue to function properly even if the manholes and duct banks are flooded. The ability of the cables to perform the function if they are flooded with sea water and the long-term effects of continued submergence in sea water is discussed in Section 8.3 of this SER.

[In response to our concern regarding internal flood protection, the applicant indicated that their discussion of plant features to prevent internal flooding of redundant safety-related equipment was in Section 6.1.3.e of the FSAR. There is no Section 6.1.3.e in the FSAR.] - 28e

[The applicant has not addressed our concern associated with the structural integrity of the safety-related structures during the design basis flood and the effects of "floating" missiles. Since the Delaware River is a navigable waterway with the refineries and naval shipyard in Philadephia, the applicant must address the effects of ships and boats with a draft of less than 12 feet hitting the walls and penetrations of safety-related structures. Some ships which do travel up and down the Delaware River and can have a draft of less than 12 feet are the "Newport" class LSTs (LST-1179 series), the "DeSoto County" class LSTs (LST-1173 series), the "Anchorage" class LSDs (LSD-36 series), submaries (especially the non-nuclear power submaries), tug boats, visiting "American" ships from foreign countries, oil tankers (when they are empty), and a large host of pleasure craft.] - 28f

Item No. 28 (Cont'd)

Because the applicant has not adequately addressed the staff's concerns identified above, we cannot conclude compliance with General Design Criterion 2 and the guidelines of Regulatory Guides 1.102, "Flood Protection for Nuclear Power Plants," Positions C.1 and 1.59, "Design Basis Floods for Nuclear Power Plants", Positions C.1 and C.2 and Branch Technical Position ASB 3-1, "Protection Against Piping Failures in Fluid systems Outside Containment". We will report resolution of these items in a supplement to this SER. The design of the facility for providing protection from flooding does not meet the acceptance criteria of SRP Section 3.4.1.

RESPONSE

a. The requested information with respect to waterproofing all safety-related structures to a higher elevation than the flood elevation of the design basis flood (PMH) has been provided in response to Question 240.8.

b. Doors 3331B and 3315B are watertight (submarine) doors and although they are installed in an unseated position (they swing inward), both doors have been designed for specified unseating pressure of 19 feet of water. To assure that these doors will not be inadvertently opened or left open, both doors are locked closed and administratively controlled during a flood event.

c. HCGS procedure "Acts of Nature", will commit to ensure that exterior doors and hatches are closed and locked by administrative procedure under impending flood conditions. Add Insert 1

d. The response to FSAR Question 410.7 has been revised to state that the site drainage system is not required to prevent the contents of failed tanks (as the result of a safe shutdown earthquake) from flooding the safety-related structures.

e. The response to NRC Question 410.9 has been revised to refer to Section 3.6.1.e instead of 6.1.3.e.

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In summary, the "Acts of Nature" procedure specifies an immediate check of all external doors to insure they are in the locked closed position upon receipt of a hurricane warning from the National Weather Service which may impact Artificial Island. The doors will be checked once per shift to verify they remain locked closed during the hurricane period unless the river level reaches site grade, at which time the doors will be checked every 30 minutes.

QUESTION 410.7 (SECTION 3.4.1)

For these nonseismic Category I vessels, pipes and tanks located outside of buildings, discuss the effect of failure of these items and any potential flooding of safety-related structures, systems and components. Provide a similar discussion for nontornado protected vessels, tanks and piping.

RESPONSE

The failure of non-Seismic Category I and non-tornado protected tanks, vessels, and major pipes located outside of buildings (Table 410.7-1) will not adversely affect safety-related structures, systems and components by flooding, as discussed below:

Failure of Tanks

The locations of tanks in the yard area are shown on Figure 1.2-1. Failure of the condensate storage tank, located on the south side of the power block (Table 410.7-1, Item 1), will not cause flooding. Any spillage due to failure of this tank will be contained within a reinforced concrete dike designed to be Seismic Category I, as discussed in Section 3.8.4.1.6.

The tanks located on the north and west sides of the power block (Table 410.7-1, Items 2 through 7) do not have Seismic Category I dikes around them. Failure of these tanks could cause local flooding. However, this flooding would not adversely affect safety-related facilities for the following reasons:

"insert A"

a. The storm drainage system in this area will drain the spillage to the Delaware River before it reaches the power plant complex.

b. Seismic Category I electrical cables and duct banks located in the vicinity of these tanks are protected against flooding, as discussed in the response to Question 410.8.

Failure of Cooling Tower Basin Wall (Table 410.7-1, Item 8)

The failure of the cooling tower basin wall would not adversely affect safety-related structures, systems and components, as discussed below:

The operating water level within the cooling tower basin is elevation 102.5 feet. The slabs and walls are conservatively designed for 3 feet of freeboard, allowing the water level to rise to elevation 105.5 feet. The grade around the basin well is

"Insert A"

a. Any spillage will be conveyed to the Delaware River by means of overland surface runoff without adversely affecting any safety-related structures, systems or components by flooding. There is a clear path to the river from the building which will assure that any surface water will not enter the building. In addition, storm drainage is provided to facilitate conveyance of runoff to the river which will further minimize the potential for any local ponding.

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at elevation 104.5 which is 2 feet above the operating water level in the basin.

The worst case flooding could result from the unlikely "wash-off" of the soil on the south side of the tower. For this case, the run-off would be dispersed and intercepted by the storm drainage system before it could reach the power block area. The Seismic Category I duct banks located between the intake structure and the power block will not be affected as they are not located in the flow path of the water.

Failure of Circulating Water Pipes (Table 410.7, Item 9)

Failure of these pipes within the yard area between the cooling tower basin and the turbine building will cause flooding of this area. Water from the damaged pipes will erode the soil cover and flood the yard. No Seismic Category I equipment or components are located in this area of possible erosion. The storm drainage system would eventually drain the water to the Delaware River.

In the most severe case, all the water from the cooling tower basin could drain through the damaged pipe into the yard area between the circulating water pumphouse and the turbine building. This could cause flooding of the lower level of the turbine building. However, safety-related systems and components would not be damaged, as discussed in the response to Question 410.115.

Failure of Major Yard Piping

Failure of any of the pipes identified in Table 410.7-1, Items 10 to 14, may cause local flooding. However, the intensity and volume of water discharge from any of these pipes is less than that of the circulating water pipes discussed above and would not cause damage to any safety-related facilities. Soil erosion caused by failure of these pipes is discussed in the response to Question 410.64.

or the water would flow overland to the Delaware River as discussed for tanks (Items 2 thru 7)

TABLE 410.7-1
YARD TANKS AND MAJOR PIPING (NON-SEISMIC)

10/83

Item No.	Tank of Pipe Description	Capacity or Flow	Location	Type of Containment	Tornado Protection
1	Condensate Storage Tank	500,000 gal	South of power plant complex	Seismic Cat. I Reinforced Conc. walls	None
2	Fire Water Tanks (2)	300,000 gal ea	North of power plant complex	None	None
3	Asphalt Storage Tank	9,000 gal	North of power plant complex	Concrete unit Masonry walls	None
4	Fuel Oil Day Tank	18,000 gal	North of power plant complex	Reinforced Conc. walls	None
5	Chemical Treatment Tanks				
	2 Sodium Hypochlorite	30,000 gal ea	North of power plant complex	Reinforced	None
	1 Sulfuric Acid	20,000 gal	North of power plant complex	Concrete	None
	2 Sodium Hypochlorite	15,000 gal ea	West of power plant complex	Walls	None
6	Sewage Treatment Plant				
	1 Equalization Tank	20,000 gal	North of power plant complex	Buried	None
	2 Treatment Tanks	8,000 gal ea	North of power plant complex	Buried	None
	1 Treatment Tank	35,000 gal	North of power plant complex	Earth berm	None
7	Fuel Oil Storage Tank	1,000,000 gal	North of power plant complex	Earth dike	None
8	Cooling Tower Basin	6,500,000 gal	North of power plant complex	Reinforced Conc. wall	None
9	144"Ø Circulating Water Pressure Pipes (2)	552,000 gpm	Between cooling tower and turbine building	Underground	Soil cover
10	48"Ø Makeup Water Pressure Pipe	30,000 gpm	Reactor building to cooling tower	Underground	Soil cover
11	36"Ø Makeup Water Pressure Pipe	21,000 gpm	Reactor building to cooling tower	Underground	Soil cover
12	48"Ø Blowdown Water Gravity Pipe	15,400 gpm	Cooling tower to Delaware River	Underground	Soil cover
13	36"Ø Deicing Water Pressure Pipe	12,000 gpm	Circulating water pipe to intake structure	Underground	Soil cover
14	12"Ø Fire Water Loop	2,500 gpm	Around plant complex	Underground	Soil cover

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DSER Open Item No. 110 A & B (Section 4.6)

FUNCTIONAL DESIGN OF REACTIVITY CONTROL SYSTEMS

The control rod drive system was reviewed in accordance with Section 4.6 of the Standard Review Plan (SRP), NUREG-0800. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the control rod drive system with respect to the applicable regulations of 10 CFR 50.

The applicant has not addressed the recommendations of NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping."

The design does not utilize a CRDS return line to the reactor pressure vessel. In accordance with NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drives Return Line Nozzle Cracking," dated November 1980, equalizing valves are installed between the cooling water header and exhaust water header, the flow stabilizer loop is routed to the cooling water header, and both the exhaust header and flow stablizer loop are stainless steel piping.

We have reviewed the extent of conformance of the Scram Discharge Volume (SDV) design with the NRC generic study, "BWR Scram Discharge System Safety Evaluation," dated December 1, 1980. The design provides two separate SDV headers, with an integral instrumented volume (IV) at the end of each header, thus providing close hydraulic coupling. Each IV has redundant and diverse level instrumentation (float sensing and pressure sensing) for the scram function attached directly to the IV. Vent and drain lines are completely separated and contain redundant vent and drain valves with position indication provided in the main control room. With respect to Design Criterion 8, the applicant stated that the "SDV Piping is continuously sloped from its high point to its low point." In order to provide a response to Design Criterion 8, the applicant must provide a description of the SDV from the beginning of the SDV to the IV drain. The description should include piping geometry (i.e., pitch, line size, orientation).

DSER Open Item No. 110 A & B (Section 4.6) (Continued)

Except for Design Criterion 8, we conclude that the design of the SDV fully meets the requirements of the above referenced NRC generic SER and is therefore acceptable. Additionally, the above-described design of the SDV satisfies LRG-II, Item 1-ASB, "BWR Scram Discharge Volume Modifications."

Based on our review, we conclude that the functional design of the reactivity control system meets the requirements of General Design Criteria 23, 25, 26, 27, 28, and 29 with respect to demonstrating the ability to reliably control reactivity changes under normal operation, anticipated operational occurrences and accident conditions including single failures, and the guidelines of NUREG-0619 and is, therefore, acceptable. We cannot conclude compliance with the guidelines of NUREG-0803 and the generic document dated December 1, 1980. The functional design of the reactivity control sytem does not meet the applicable acceptance Criteria of SRP 4.6. We will report resolution of these items in a supplement to this SER.

RESPONSE

The concerns of NUREG-0803 are addressed in response to Q410.26.

FSAR Section 4.6.1.2.4.2(f) has been revised to include a description of the SDV piping.

room. Differential pressure between the reactor vessel and the cooling water header is indicated in the main control room. Although the drives can function without cooling water, seal life is shortened by long-term exposure to reactor temperatures. The temperature of each drive is indicated and recorded, and excessive temperatures are annunciated in the main control room.

e. Exhaust water header - The exhaust water header connects to each HCU and provides a low pressure plenum and discharge path for the fluid expelled from the drives during control rod insert and withdraw operations. The fluid injected into the exhaust water header during rod movements is discharged back up to the RPV via reverse flow through the insert exhaust directional solenoid valves of adjoining HCUs. The pressure in the exhaust water header is, therefore, maintained at essentially reactor pressure. To ensure that the pressure in the exhaust water header is maintained near reactor pressure during the period of vessel pressurization, redundant pressure equalizing valves connect the exhaust water header to the cooling water header.

f. Scram discharge volume - The scram discharge volume (SDV) consists of two sets of 12 inch diameter header piping, each of which connects to one-half of the HCUs and drains into a 12 inch diameter scram discharge instrument volume (SDIV). Each set of header piping is sized to receive and contain all the water discharged by one-half of the drives during a scram, independent of the SDIV.
The header piping slopes to a low point with a minimum pitch of 1/8" per foot as shown on Figure 4.6-1
The SDIV for each header set is directly connected to the low point of the header piping. The large-diameter pipe of each SDIV thus serves as a vertical extension of the SDV. A 2" piping connection at the bottom of the SDIV provides drainage of the SDIV and SDV via sloped drain lines with a minimum 1/8" per foot slope.
During normal plant operation, the SDV is empty and is vented to the atmosphere through its open vent and drain valves. When a scram occurs, upon a signal from the safety circuit, these vent and drain valves are closed to conserve reactor water. Redundant vent and drain valves are provided to ensure against loss of reactor coolant from the SDV following a scram. Lights in the main control room indicate the position of these valves.

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QUESTION 410.26 (SECTION 4.6)

Provide the information requested in our generic letter 81-34, dated August 31, 1981, regarding NUREG-0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping."

RESPONSE

HCGS is participating in the BWROG activities related to the scram discharge pipe integrity. The BWROG's final response to the NRC is being prepared for NRC review and approval. ~~A HCGS plant specific response will be provide in June 1984~~.
A HCGS plant specific response will be provided within 60 days of NRC resolution of the BWROG position. HCGS will implement any required fix by the end of the next refueling outage which is at least 12 months after NRC resolution. Pending material availability, this schedule may change with NRC approval.

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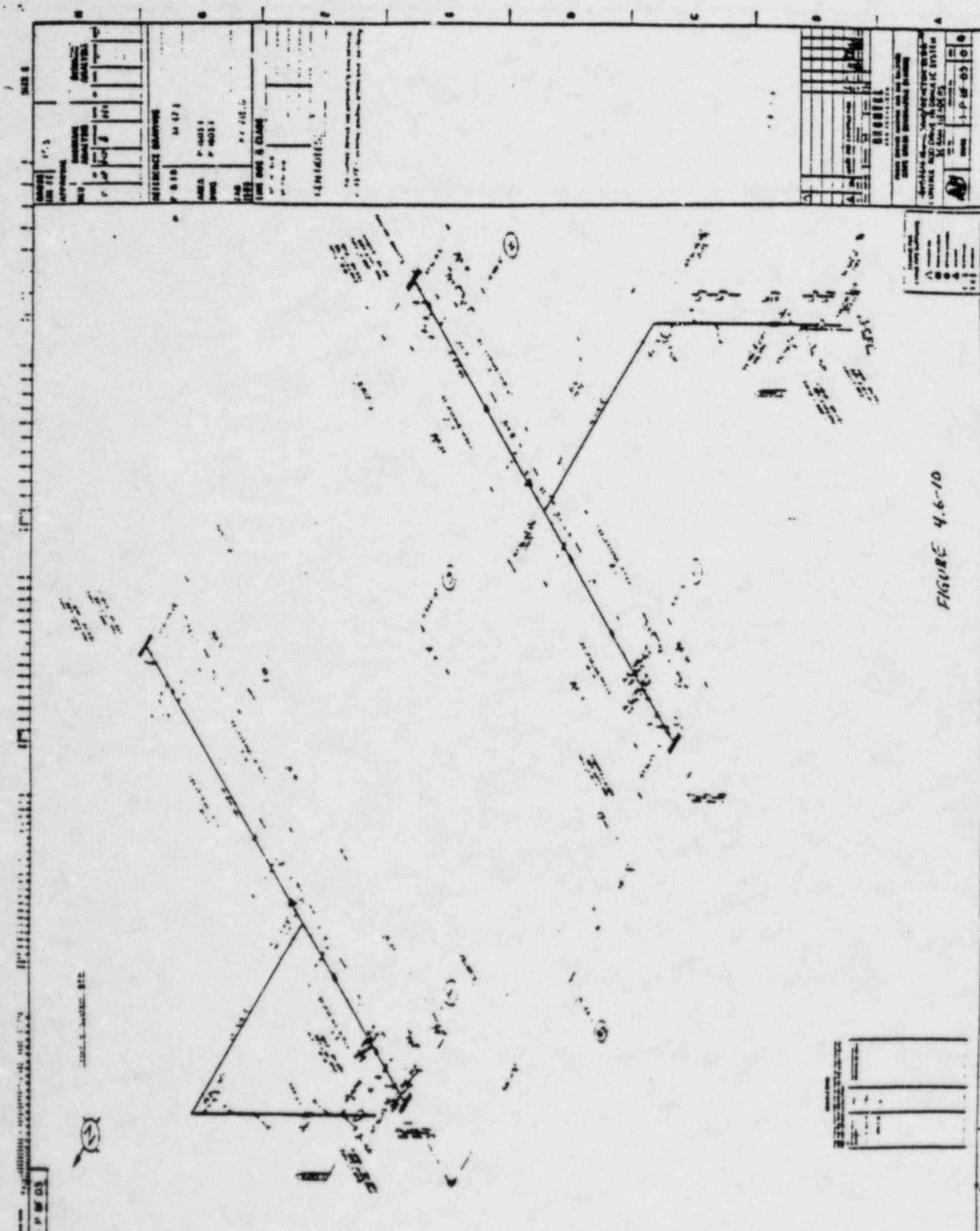

FIGURE 4.6-10

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DSER Open Item No. 112 (DSER Section 5.2.5)

REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION

Provisions have not been made to monitor all of the systems connected, as identified in Table 1 of Section 5.2.5 of the Standard Review Plan, to the RCPB for monitoring and alarming intersystem leakage by using radioactivity and differential flow monitors. Specifically, the applicant has not provided monitoring capabflity for intersystem leakage for the safety injection system (high and low pressure systems), residual heat removal system (inlet and discharge), reactor core isolation cooling system, and the steam side of the high pressure coolant injection system. Thus, the guidelines of Regulatory Guide 1.45, Position C.4 are not met. Each leakage detection system has indicators and alarms either in the control room or at the local panels. The monitor signals provided to the control room are generated through the plant computer system with no unprocessed signals available to the operators and no procedures to direct the operators where or how to obtain the information if the control room indications are lost. The applicant should provide a discussion of the capability to maintain sufficient onsite manpower at all times to man all local panels 100% of the time (this is in addition to the manpower requirements discussed in Section 9.5 of this SER) when the information is not available in the control room, to provide a seismic Category I communication system between the control room and all local panels, to provide procedures to guide the personnel at the local panels, and to propose a Technical Specification requiring the manning of the local panels when the control indications are not available. Thus, the guidelines of Regulatory Guide 1.45, Position C.7 is not met.

The applicant does not have a sump flow monitoring system, an airborne particulate radioactivity monitoring system, and a seismic Category I monitoring system and therefore does not meet the guidelines of Positions C.3 and C.6 of Regulatory Guide 1.45. As recommended by Regulatory Guide 1.45, at least three separate detection methods should be employed and two of these methods are to be (1) sump level and flow monitoring, and (2) airbone particulate radioactivity monitoring. We will require the applicant to provide sump flow monitoring, in addition to the existing sump level monitoring stated in the FSAR, in order to meet the first part of Position C.3. The applicant has not provided an airborne particulate radioactivity monitoring system. Not having an airborne particulate radioactivity monitoring system is acceptable provided that the applicant provides an alternate monitoring system which meets the qualifications of the airborne particulate system. The applicant has not proposed any alternate at this time. In conformance with Regulatory Guide 1.45, Position C.3, the third method of detecting leakage is the monitoring of drywell cooler condensate flows. Regulatory Guide 1.45, Position C.6, requires the airborne particulate monitoring system to be seismic Category I. The applicant must provide a seismic Category I airborne radioactivity monitoring system or a seismic Category I acceptable alternate leakage monitoring system.

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The applicant has not provided information concerning the systems testing and calibration frequency and capability during power operation of the plant in accordance with Regulatory Guide 1.45, Position C.8. The applicant has committed to specifying the maximum allowable identified and unidentified leakage rates as 25 gpm and 5 gpm, respectively, in the technical specifications. Thus, the guidelines of Regulatory Guide 1.45, Position C.9, are met. Until the applicant provides the information stated above on the leakage detection systems, we cannot make any conclusions as to the acceptability of the systems. We will report resolution of this item in a supplement to this SER.

RESPONSE:

For the HCGS definition of intersystem leakage, refer to Section 1.14.1.7.

For a discussion on leak detection for the four systems noted, refer to the following sections:

1. Safety Injection System (high and low pressure systems) - Section 5.2.5.2.1 (o).
2. Residual Heat Removal System (inlet and discharge) - Section 5.2.5.2.1 (o).
3. Reactor Core Isolation Cooling System - Section 5.2.5.2.1 (m)
4. High Pressure Coolant Injection System (steam side) - Section 5.2.5-2.1 (l).

Section 5.2.5.2 has been revised to indicate that the drywell floor and equipment drain sump leakage rate indications are class 1E and are located on main control room panel 10C604.

Sections 1.8.1.45 and 5.2.5.2 have been revised to address the concerns of positions C.3 and C.6 of Regulatory Guide 1.45.

Section 5.1.5.2 has been revised to identify that the drywell equipment and floor drain sump level monitoring instrumentation is seismic Category I.

Sections 5.2.5.9 and 11.5.2.2.15 have been revised to provide information concerning testability.

Note: No changes to this sheet. Included for clarification only.

See Section 5.2.3 and 6.1 for further discussion and Section 1.8.2 for the NSSS assessment of this Regulatory Guide.

1.8.1.45 Conformance to Regulatory Guide 1.45, Revision 0 May 1973: Reactor Coolant Pressure Boundary Leakage Detection Systems

HCGS is designed to comply with Regulatory Guide 1.45, with the exceptions, clarifications, and amplifications discussed below.

Paragraph C.3 of Regulatory Guide 1.45 requires that three methods of leak detection be provided. HCGS does not employ an airborne particulate radioactivity monitor due to uncertainties in detecting 1 gpm of RCPB leakage in 1 hour. The uncertainties that affect the reliability, sensitivity, and response times of radiation monitors, especially iodine and particulate monitors, are discussed below.

The amount of activity becoming airborne following a 1-gpm leakage from the RCPB varies, depending upon the leak location and the coolant temperature and pressure, which affect the flashing fraction and partition factor for iodines and particulates. Thus, an airborne concentration cannot be correlated to a quantity of leakage without knowing the source of the leakage.

Coolant concentrations during operation can vary by as much as several orders of magnitude within several hours. These effects are mainly due to spiking during power transients or changes in the use of the reactor water cleanup (RWCU) system. An increase in the coolant concentrations can give increased containment concentrations when no increase in unidentified leakage occurs.

Not all activity is from unidentified leakage. Changes in other sources result in changes in the containment airborne concentrations. For example, identified leakage is piped to the drywell equipment drain sump, but all sump and collection drains are vented to the drywell atmosphere, thereby allowing particulates to escape, causing further measurement uncertainties.

The amount of activity that is detected depends upon the amount of plateout on drywell surfaces prior to reaching the detector intake. The amount of plateout is dependent on uncertain

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quantities, such as location of the leak, distance from the detectors, and the pathway to the detector.

Furthermore, under normal operating conditions a radiation-free background does not exist. There is a buildup of activity concentration due to both identified and unidentified leakage. At high equilibrium activity levels, a small change in activity level due to a small leak is hard to detect in the desired time interval.

Although particulate monitors are available with sensitivities covering concentrations expected in the drywell, previously discussed uncertainties under operating conditions coupled with any calibration and setpoint uncertainties make particulate monitors a less reliable method of leak detection.

HCGS does employ five ~~three~~ separate and diverse leak detection methods. The RCPB leak detection system consists of:

a. SEISMIC CATEGORY I QUALIFIED ~~D~~drywell floor and equipment drain sump level monitors (IN LIEU OF A SEISMIC CATEGORY I AIR PARTICULATE DETECTION SYSTEM).

b. A drywell cooler condensate flow monitor

c. A noble gas monitor (IN LIEU OF AN AIR PARTICULATE DETECTION SYSTEM

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Paragraphs C.2 and 5 require that the leakage monitors be able to detect an increase in leakage of 1 gpm in 1 hour. The noble gas monitor can detect concentrations as low as 10-[illegible] µCi/cc, the minimum activity concentration expected in the drywell based on the primary system coolant. However, an increase in 1 gpm leakage within an hour may be difficult to detect due to high equilibrium activity levels for noble gases (10-[illegible] to 10-[illegible] µCi/cc) and buildup of background radiation. The noble gas monitor is capable of detecting leaks of approximately 10 gpm and does so very quickly due to the high diffusion rates of the noble gases.

The drywell floor drain sump level monitor and the drywell cooler condensate monitor can detect fluid flows of 1 gpm in 1 hour. However, fluid flow is not always a direct indication of RCPB leakage because of free communication between the suppression chamber and the drywell. The drywell atmosphere is not necessarily saturated due to the water vapor removal by the drywell coolers. Hot water can evaporate from the torus and

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d. Seismic Category I drywell pressure monitors

e. Seismic Category I drywell temperature monitors.

Leakage flows into the drywell floor and equipment drain sumps are not measured directly due to physical configuration which makes it impractical to do so. As stated in Section 5.2.5.2, leakage flow into the sumps is calculated based on the rate of change of level in the sumps.

Sump pump starts and stops and duration of pumpout are monitored by the Class 1E radiation processor. An alarm is annunciated in the main control room whenever pumpout duration exceeds a predetermined time limit. Total sump pumpout can be calculated based on the duration of pumpout and the constant known flowrate of the sump pump provided that only one pump is required to lower the sump level. The starting of the second pump is a positive indication of excessive leakage into the sump or is an indication that the first pump has failed with either event requiring operator action. The high-high level condition which initiated the operation of the second pump is annunciated in the main control room.

enter the drywell. The water will condense and register on the drywell cooler condensate monitor. The condensate drains into the drywell floor drain sump and will register on the sump level monitor. Therefore, during times of suppression pool transients, such as from heat up from main steam safety/relief valve (SRV) or HPCI system testing, evaporation from the suppression chamber will obscure values of RCPB leakage.

~~Position C.6 requires that the leakage detection systems be capable of performing their functions after a seismic event that does not require plant shutdown. The leak detection system is capable of operating after an operating basis earthquake (OBE) and a DBA. The sump level monitor is used for both Regulatory Guides 1.45 and 1.97 purposes.~~

~~Position C.6 also suggests that at least one RCPB leak detection method should remain functional after an SSE. This capability does not exist in the HCGS design. The purpose of the RCPB leak detection system is to monitor the integrity of the RCPB so that if there are any changes, the plant can be safely shut down. Since the plant will shut down after an SSE, the leak detection system does not have to remain functional after an SSE, should it occur.~~

Position C.7 requires that indicators and alarms for each leakage detection system should be provided in the main control room. Procedures for converting various indications to a common leakage equivalent should be available to the operators. The calibration of the indicators should account for needed independent variables.

Position C.7 is further clarified by Standard Review Plan Section 5.2.5, III.5 which requires that if monitoring is computerized, backup procedures should be available to the operator.

— INSERT A —

~~In the drywell sumps and drywell air coolers leakage monitoring systems, level and level change is electronically transmitted from level sensors to a local radiation processor (LRP) which processes these signals and in turn transmits processed data for indication and alarms, levels, and calculated flow rates to the central radiation processor (CRP) in the computer room. Data in the CRP is available to the operator on the display keyboard printer (DKP) terminal CRT and/or annunciated in the main control room.~~

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The drywell air coolers' leakage monitoring and noble gas monitoring systems' signals are processed by local radiation processors which then transmit the processed data to the main control room via the central radiation processor (CRP). The CRP in turn makes this indicating and alarming information available to the control room operator via CRT displays.

These signals are processed locally by local radiation processors (LRPs) which are provided with digital readout indicators. These indicators provide information to the operator in the same format (using the same engineering units) as the information provided by the CRP through the CRTs in the main control room. Since these indications are of the same format, procedures for converting the LRP indication to a common leakage equivalent (to that normally provided in the main control room) are unnecessary.

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Since these ~~the~~ leakage signals are processed locally with capability for local readout, procedures for converting various indications to a common leakage equivalent are not provided to the operators, nor are ~~Backup~~ backup procedures ~~are not~~ provided to the operator ~~, nor are unprocessed signal indications provided in the main control room.~~

— INSERT B —

~~However, all processed data can always be read at the LRP if the CRP and DKP become unavailable.~~

~~Position C.8 requires that leakage detection systems be equipped to readily permit testing for operability and calibration during plant operation. This capability is not provided on RCPB leak detection instrumentation inside the primary containment, because calibration and testing cannot be performed inside the containment during reactor operation.~~

For further discussion of the RCPB leak detection system, see Section 5.2.5.

1.8.1.46 Conformance to Regulatory Guide 1.46, Revision 0, May 1973: Protection Against Pipe Whip Inside Containment

The criteria set forth in Regulatory Guide 1.46 are design bases for HCGS. See Section 3.6.2 for further discussion of pipe break design and Section 1.8.2 for the NSSS assessment of this Regulatory Guide.

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As described in Section 5.2.5.2, displays of drywell equipment and floor drain sump levels (which are not dependent of the non-1E plant computer systems) are provided on panel 10C604 in the main control room.

Position C.8 requires that the leakage detection systems should be equipped with provisions to readily permit testing for operability and calibration during plant operation. This is interpreted to mean channel functional testing as defined in the technical specifications (Chapter 16). Calibration of the leakage detection systems is performed during plant outages per the technical specifications. Calibration of the drywell floor and equipment drain sump level monitoring systems can not be performed at power due to the fact that the sensors are located inside the drywell and are therefore inaccessible during power operation. Rosemount 1153 transmitters are used throughout the plant and are typically calibrated on an 18 month cycle (reference NUREG-0123). This model transmitter is used for the sump level transmitter. In addition, the calibration accuracy of these transmitters can be observed on an ongoing basis by comparing the level readings with known independently measured sump levels at which the sump pumps start or stop. The pumps are started and stopped using electro-mechanical float switches. It should also be noted that the rate of change readings (sump inflow) obtained from these transmitters will be substantially free from the effects of drift due to the sampling frequency. The sensors for the drywell cooler condensate flow monitoring systems and the drywell temperature monitoring system are also located inside the drywell (and therefore inaccessible during power operation). However, these sensors are RTDs and access to them for normal instrument channel calibration is not required. The remaining leak detection monitoring systems discussed above have the capability of being calibrated during operation.

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the RWCU pump heat exchangers and the reactor recirculation pump seal and jacket cooling heat exchangers. The RACS sensor monitors radiation emanating from a continuously flowing RACS water sample which is taken at a point downstream of the RACS pumps.

High radiation in the SACS water or the RACS water indicates intersystem leakage. The affected sensor and its associated monitoring channel will activate an alarm in the main control room when the radiation exceeds a predetermined limit. No isolation trip functions are performed by these channels.

These radiation channels are part of the process radiation monitoring system described in Section 11.5.

High levels in the SACS or RACS head tanks may also indicate intersystem leakages from the sources given above. High level in either head tank will activate an alarm in the main control room.

5.2.5.2 Leak Detection Instrumentation and Monitoring

5.2.5.2.1 Leak Detection Instrumentation and Monitoring Inside Primary Containment

a. Floor drain sump level and flow - The normal design leakage collected in the floor drain sump includes unidentified leakage from the control rod drives (CRDs), valve flange leakage, component cooling water, service water, air cooler drains, and any leakage not connected to the equipment drain sump.

— INSERT C —

~~A level transmitter is used in the drywell floor drain sumps and is fed into a local microprocessor. A level change in the sump will be converted to flow rate by the processor. Abnormal leakage rates are alarmed in the main control room. Collection in excess of background leakage would indicate an increase in reactor coolant leakage from an unidentified source in excess of 1 gpm within 1 hour.~~

b. Equipment drain sump level and flow - The equipment drain sump collects only identified leakage and valve stem packing leakoff collectively. This sump receives

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A Class 1E level transmitter is used to monitor the drywell floor drain sump with the level signal being supplied to a Class 1E radiation processor of the Class 1E radiation monitoring system (RMS) (panel 10C604) located in the main control room. A level change in the sump is converted to a flow rate by the processor and leakage rates can be displayed continuously at panel 10C604 and are available, via data link, at the operator's console CRT. An increase in unidentified leakage in excess of technical specification limits is alarmed in the main control room.

The floor drain sump level monitoring instrumentation is qualified to remain functional following a safe shutdown earthquake (SSE)

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piped drainage from pump seal leakoff and reactor vessel head flange vent drainage. The equipment drain sump instrumentation is identical to the floor drain sump instrumentation.

c. Drywell air cooler condensate drain flow - Condensate from the drywell air cooler is routed to the floor drain sump.

~~Condensate in each of two drain lines from the eight drywell air coolers drains into a line and is trapped by a closing solenoid valve controlled by a local microprocessor. The rising level in the drain line is sensed by a level transmitter that sends a signal to~~

Flow in each of the two drain headers from the eight drywell coolers (four coolers per header) is monitored by a flow sensor. The flow signal from each flow sensor is processed by a local radiation processor which transmits the flow data to the main control room, via the central radiation processor, for indicating and alarm functions. Any flowrate increase exceeding technical specification limits will be alarmed in the main control room.

This flow monitoring instrumentation is capable of operation following seismic events which do not require plant shutdown.

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to differentiate between identified and unidentified leakage is discussed in Sections 5.2.5.4, 5.2.5.5, and 7.6.

5.2.5.7 Sensitivity and Operability Tests

Sensitivity, including sensitivity testing and response time of the leak detection system, and the criteria for shutdown if leakage limits are exceeded, is covered in Section 7.6.

Testability of the leakage detection system is contained in Section 7.6.

5.2.5.8 Safety Interfaces

The Balance of Plant-GE Nuclear Steam Supply System (NSSS) safety interfaces for the leak detection system are the signals from the monitored balance of the plant equipment and systems that are part of the nuclear system process barrier, and associated wiring and cable lying outside the NSSS equipment.

5.2.5.9 Testing and Calibration

~~Provisions for testing and calibration of the leak detection system are covered in Chapter 14.0.~~

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5.2.5.10 Conformance to Regulatory Guide 1.45

For a discussion of compliance with Regulatory Guide 1.45, see Section 1.8.1.45.

5.2.5.11 SRP Rule Review

SRP 5.2.5 acceptance criterion II.1 requires that leak detection system integrity must be maintained following an earthquake, as per GDC2. This is met through Regulatory Guide 1.29 positions C-1 and C-2.

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Testing and calibration will be in conformance with the Technical Specifications and will consist of channel checks and channel functional tests during power operation. Channel calibration will be done during refueling outages.

Testing and calibration of the noble gas monitor is discussed in Section 11.5.2.15.

Note: No changes to this Sheet. Included for clarification only.

information about the HEPA and charcoal filter efficiency and condition.

11.5.2.2.12 Radwaste Area Exhaust Radiation Monitoring System

The RAE RMS is located in the exhaust duct for radwaste area compartments in which there is equipment that has a possibility of releasing airborne radioactive materials (Refer to Figure 11.5-1). The RAE RMS is upstream of the filters and will be exposed to higher concentrations than the RES RMS, thus allowing earlier detection of any problems in the radwaste areas of the auxiliary building. The RAE RMS has the same components and functions as the RBVSE RMS described in Section 11.5.2.2.8.

11.5.2.2.13 Gaseous Radwaste Area Exhaust Radiation Monitoring System

The gaseous radwaste area exhaust (GRAE) RMS is located in the exhaust duct for the recombiner compartments (Refer to Figure 11.5-1). This allows earlier detection of airborne radioactive materials than is possible by downstream monitors where the concentrations are more diluted. The GRAE RMS has the same components and functions as the RBVSE RMS described in Section 11.5.2.2.8. There are no filters upstream of the location.

11.5.2.2.14 Technical Support Center Ventilation Radiation Monitoring System

The technical support center ventilation (TSCV) RMS is located in the inlet plenum for the technical support center (Refer to Figure 11.5-1) The purpose of the TSCV RMS is to detect radioactive materials in the inlet air. The TSCV RMS has the same components as the RBVSE RMS described in Section 11.5.2.2.8. If the concentration exceeds the trip setpoint, an alarm at the CRP alerts the operator to manually transfer from the normal air supply to an emergency recirculation and filtration mode.

11.5.2.2.15 Drywell Leak Detection Radiation Monitoring System

The drywell leak detection (DLD) RMS monitors the gaseous radioactive materials in the drywell (Refer to Figure 11.5-3). The design objective of this system is to monitor reactor coolant pressure boundary (RCPB) leakage in accordance with Regulatory Guide 1.45. Conformance to Regulatory Guide 1.45 is discussed in Section 1.8. The capability to do so declines as the normal in-containment background of gaseous radioactive materials increases because of the accumulation from identified leaks. An air sample is extracted and returned through penetrations that are isolated by the PCIS described in Section 7.3.1.1.5. The DLD RMS components are one inlet and one outlet stub on the east side of the drywell, penetrations, and isolation valves. There is also a shield sample chamber, a beta scintillation detector, and an LRP. The high-high alarm indicates excessive leakage from the RCPB. The DLD RMS is seismically qualified to operate under conditions during which the reactor is operated. The functional requirements and descriptions of other leak detection equipment are discussed in Sections 5.2.5 and 7.6.1.3. Provision for a grab sample is included.

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11.5.2.2.16 Reactor Auxiliaries Cooling System Radiation Monitoring System

The reactor auxiliaries cooling system (RACS) RMS monitors a sample extracted from the RACS (Refer to Figure 11.5-1). The RACS RMS has the same components as the liquid radwaste RMS. The high-high alarm indicates leakage into the RACS from the heat exchangers that are serviced by the RACS.

11.5.2.2.17 Safety Auxiliaries Cooling System Radiation Monitoring System

The safety auxiliaries cooling system (SACS) RMS has two monitors, A and B, one for each of the two SACS loops (Refer to Figure 11.5-1). The SACS RMS monitor samples extracted from the SACS. The SACS RMS has the liquid radwaste RMS. The SACS RMS sample chambers are part of the SACS pressure boundary and are seismically qualified. The high-high alarm indicates leakage into the SACS heat exchangers from the safety auxiliaries served by the safety auxiliaries cooling system.

11.5.2.2.18 Heating Steam Condensate, Waste Radiation Monitoring System

The heating steam condensate, waste (HSCW) RMS monitors a sample of the condensate flow from the liquid waste management system (Refer to Figure 11.2-4). The high-high alarm/trip indicates both leakage of radioactive materials from one or both of the

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Testing and calibration of the DLD RMS will be in conformance with the Technical Specifications and will consist of channel checks and channel functional tests during power operation. Channel calibration will be done during refueling outages.

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DSER Open Item No. 141 (DSER Section 9.1.3)

SPENT FUEL POOL COOLING AND CLEANUP SYSTEM

The applicant has not provided a discussion of the means to provide cooling to the spent fuel pool after a safe shutdown earthquake which fails the non-seismic Category I skimmer tanks in such a manner as to plug the tank drains. Therefore, we cannot conclude that this design satisfies the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guides 1.13, "Spent Fuel Storage Facility Design Basis," Positions C.1, C.7 and C.8, and 1.29, "Seismic Design Classification," Positions C.1 and C.2.

The applicant has not adequately addressed the concern of high- and moderate-energy piping system failures and the means to protect these systems (refer to Section 3.6.1 of this SER.) Thus, we cannot conclude that the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," and the guidelines of Regulatory Guide 1.13, Positions C.2, are satisfied.

The system is accessible for routine visual inspection of the system components. Both fuel pool cooling pumps are required to operate at all times to remove the maximum normal heat load. Thus, the cooling system does not meet the single failure criterion. The applicant has not committed to include the portions of the cooling and cleanup systems which are not normally operating in the inservice inspection and periodic functional testing programs as described in Sections 6.6 and 3.6.6 of the SRP. The applicant has not specified the frequency of the testing. Thus, the requirements of General Design Criterion 45, "Inspection of Cooling Water Systems," and 46, "Testing of Cooling Water Systems," are not satisfied.

The spent fuel pool cooling system will maintain the fuel pool water temperature at 135°F, with a heat load of 16.0 MBtu/hour based on decay heat generation from 3,668 fuel bundles (maximum storage) and both cooling trains in operation. This is the normal discharge from 15 fuel cycles. The spent fuel pool cooling system consists of two pumps with a common suction line and a common discharge line, which feeds two heat exchangers with a common inlet line and a common outlet line. Each pump and each heat exchanger have a manual isolation valve on the inlet and manual isolation valve on the outlet; thus, each component can be independently isolated. If one pump or one pump and heat exchanger were not available under these conditions, the pool temperature would exceed the 140°F specified in the Standard Review Plan. The pool cooling must maintain a pool temperature of less than 140°F with any single active failure.

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However, the full flow by-pass line around the non-safety-related cleanup water system has not been clearly indicated in the FSAR figures. Therefore, we cannot conclude that the requirements of General Design Criterion 44, "Cooling Water," are met.

Until the applicant provides acceptable responses, we cannot conclude that the system conforms to the requirements of General Design Criteria 2, 4, 44, 45, and 46 as they relate to protection from natural phenomena, missile and environmental effects, cooling water capability, inservice inspection, and functional testing and the guidelines of Regulatory Guides 1.13, Positions C.1, C.2, C.7, and C.8, 1.29, Positions C.1 and C.2 relating to the system's functional design and seismic classification. The spent fuel pool cooling and cleanup system does not meet the applicable acceptance criteria of SRP-9.1.3. We will report resolution of this item in a supplement to this SER.

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

See the revised response to FSAR Question 410.55 and revised Section 9.1.3.3 for a discussion of the seismic response of the skimmer surge tanks.

Section 3.6 describes the method of protection against dynamic effects associated with postulated ruptures in high and moderate energy piping located both inside and outside the primary containment. The FPCC and Torus Water Cleanup Systems are classified as moderate energy systems. The failure of high and other moderate energy piping on FPCC and torus water cleanup systems has been evaluated in Section 3.6. Because of the physical separation of the FPCC and torus water cleanup systems from high and other moderate energy piping, it has been concluded that a postulated piping failure in high and/or other moderate energy piping will not adversely affect the operation of these systems. Therefore,

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There are no high or moderate energy lines above or near the spent fuel pool whose failure would adversely affect the spent fuel pool or the storage racks. Piping within the spent fuel pool is seismically designed moderate energy piping. A crack in this piping would not have an adverse effect on either the spent fuel pool or the storage racks.

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it can be concluded that the systems design meets the requirements of GDC No. 4, "Environmental and Missile Design Bases", and the guidelines of Regulatory Guide 1.13, Position C.2. For discussion on moderate energy leakage in the common spent fuel pool cooling pump discharge line, see response to Question 410.48.

The spent fuel cooling system does not perform a specific function in shutting down the reactor or in mitigating the consequences of an accident; therefore, does not meet the criteria for being included in ASME B&PV Code Section XI testing requirements. ADD INSERT B

As discussed below, there is no single active failure within the FPCC system which will result in the loss of a FPCC heat exchanger. However, two system configurations (one FPCC Pump and two FPCC heat exchangers and one FPCC pump and one FPCC heat exchanger) have been evaluated as requested. The results are provided in Table 141-1.

The evaluation indicates that in the event of a single active failure of one FPCC pump, the spent fuel pool temperature could reach 152°F, which exceeds the SRP 9.1.3 limit of 140°F. It is conservatively estimated that the fuel pool temperature could exceed 140°F for 26 days under these conditions. This is based on worst-case assumptions. A maximum SACS water temperature of 95°F is assumed. In addition, a maximum accumulation of spent fuel is assumed stored in the fuel pool, i.e., 16 consecutive refuelings at 18 month intervals, to fill the high density racks to their maximum capacity of 3984 spent fuel assemblies (which exceeds the SRP 9.1.3 requirements). It is also assumed that the last 1/3 core is placed in the spent fuel pool as quickly as practical after shutdown, i.e. 8 days. This is slightly longer than the 150 hours recommended by SRP 9.1.3 and is based on the BWR servicing and refueling improvement program - Phase 1 Summary Report prepared by GE (NEDG-21860).

Review of Figure 9.1-5, Sheet 1 of 2, confirms that there is no single active failure mechanism within the FPCC system which will render one heat exchanger unavailable (e.g., inadvertent valve actuation.). In addition, preventive maintenance on the FPCC heat exchangers can be scheduled prior to the refueling outage to ensure the availability of both heat exchangers to remove the calculated maximum normal heat load. The plate type

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Spent fuel pool cooling and cleanup system piping will be visually inspected once every 18 months. System pumps will be start/stop tested once every 30 days if they have not been used within the previous 30 days.

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heat exchanger is a low maintenance component with long life gaskets that are expected to be replaced about every 5 years. In addition, the manufacturer has performed a reliability and maintainability analysis on the plate-type heat exchangers which indicates that failures are extremely unlikely. Therefore, failure of one FPCC heat exchanger is not considered to be a credible event. Table 141-1 also provides the maximum pool heatup rate for these postulated events. The time to reach the maximum temperature is conservatively based on a constant heatup rate.

A single active failure of one of the SACS cooling loop inlet valves to the FPCC system heat exchangers has also been evaluated. This could render the FPCC system heat exchangers unavailable for a short period of time. However, the fuel pool cooling is re-established in a short period of time by either manually re-opening the affected valve or providing cooling from the standby SACS loop. It is anticipated that the fuel pool heat-up rate during this short period will not cause the fuel pool temperature to exceed 140°F.

During normal operation, the offsite doses from the fuel pool are negligible. Elevating the fuel pool temperature to 152°F or 174°F would result in a slight increase in the evaporation rate. This slight increase would result in a slight increase in the offsite doses, however, the doses would still be negligible and well below the 10CFR20 limits.

Elevated pool temperature, up to 165°F, will not ~~significantly~~ affect the performance of the fuel pool filter demineralizer. Above 165°F, the only adverse factor is a slightly reduced capacity for ion exchange. Up to 175°F, approximately a 10% reduction in run length of the demineralization cycle is expected with no change in the filter capacity.

The response to Question 410.46 has been revised to address the failure of one FPCC pump. As stated in Section 9.1.3.1.j, normal makeup capability is provided to makeup evaporation losses and to ensure that fuel pool cooling is maintained.

FSAR Figure 9.1-5, Sheet 1 of 2, identifies the full flow by-pass lines around the non safety-related filter-demineralizer system (10"-HBC-062, 6"-HBC-062, 6"-HCC-015). This mode of operation is discussed in Section 9.1.3.2.3, and meets the requirements of General Design Criterion 44, "Cooling Water".

FSAR Section 9.1.3.2.2.4 has been revised to provide the requested information on spent fuel pool sampling.

TABLE 141-1

Single Active Failure Analysis for FPCC System

Description of Parameter	System Configuration	
	1 FPCC Pump and 2 HX	1 FPCC Pump and 1 HX
1. Normal Max. heat load	16.1×10^6 BTU/hr	16.1×10^6 BTU/hr
2. Cooling Water (SACS) Temperature	95°F	95°F
3. Maximum Fuel Pool Temperature	152°F	174°F
4. Heat-up Rate	1.02°F/hr	2.26°F/hr
5. Evaporation Rate	2.13 gpm	5.99 gpm
6. Time to reach the Maximum Temperature assuming the Fuel Pool Temperature at 135°F.	16.7 hrs	17.3 hrs

b. The FPCC system cooling loop (consisting of skimmers, surge tanks, fuel pool cooling pumps, fuel pool heat exchangers, and interconnecting loop piping) and the emergency fuel pool water makeup piping are designed to meet Seismic Category I requirements, except for the surge tanks. The surge tanks are of non-Seismic Category I design, but are embedded in a Seismic Category I concrete structure that provides the pressure boundary for this part of the FPCC system cooling loop. The FPCC system purification loop, consisting of the filter-demineralizers, their interconnecting piping, and associated equipment, is non-Seismic Category I.

c. The FPCC system is designed to handle the decay heat released by all anticipated combinations of spent fuel that could be stored in the fuel pool. The pool water temperature is maintained at a maximum of 135°F under the design load of 16.1 ~~16.0~~ x 10^6 Btu/h. This heat load is based on 16 ~~15~~ consecutive refuelings with one-third of the core removed during each refueling, and on a refueling frequency of 18 months.

d. The FPCC system is designed to permit the residual heat removal (RHR) system to be operated in parallel with the FPCC system through a crosstie, to remove the maximum heat load and to maintain the bulk water temperature in the spent fuel pool at or below 150°F, with a maximum anticipated heat load of 34.2 ~~31.0~~ x 10^6 Btu/h. This is based on one full core load of fuel at the end of a fuel cycle, plus the decay heat of the spent fuel discharged at the thirteen ~~two~~ previous refuelings. ADD Insert 1

e. The FPCC system is designed with additional capability to provide a source of makeup water to ensure against loss of fuel pool cooling, in compliance with Regulatory Guide 1.13.

f. The FPCC system is designed to monitor fuel pool water level and potential leakage paths and maintain a sufficient level above the spent fuel elements to provide radiation shielding for normal building occupancy.

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If required, one RHR pump and one RHR heat exchanger can be aligned to augment the FPCC system through the system crosstie. For this system configuration, a heat load greater than 45 million Btu/hr can be removed from the spent fuel pool with a SACS inlet temperature of 95°F and a spent fuel pool temperature of 150°F.

9.1.3.2.2.2 Fuel Pool Cooling Pumps

Two single-stage, horizontal, motor-driven, centrifugal, half-capacity recirculation pumps circulate water through the FPCC system. The pumps are piped in parallel and take suction from the skimmer surge tanks through a common header. The pump motors, pump control circuits, and power supplies are Class 1E. Each pump is provided with controls for starting and stopping the motor as follows: For normal and accident operation, the primary control in the MCR is used. If it is necessary to start or stop either pump when the MCR is inaccessible, the control in the remote shutdown panel (RSP) is used. Each pump is automatically stopped by skimmer surge tank low-low level, low suction pressure, or low discharge flow.

9.1.3.2.2.3 Fuel Pool Heat Exchangers

Two half-capacity, plate-type heat exchangers are provided for the FPCC system. They are designed to transfer the system design
16.1 heat load of ~~16.0~~x10 Btu/h from 135°F pool water, flowing at the system design flow rate of 1400 gpm, to the safety auxiliaries cooling system (SACS) at its maximum temperature of 95°F.

The heat exchangers are arranged in parallel. Fuel pool heat exchanger inlet and outlet temperatures are monitored and recorded by the control room integrated display system (CRIDS).

9.1.3.2.2.4 Fuel Pool Filter-Demineralizer System

The cleanup loop of the FPCC system includes a filter-demineralizer system located in the auxiliary building. The filter-demineralizer system consists of two vessels, located separately in shielded cells, and two holding pumps. One of the vessels, including its holding pump, normally serves as a spare. The holding pumps and the equipment common to the two vessels, including the resin tank with agitator, dust evacuator, and resin eductor, and the associated piping, valves, and instrumentation, are located in a separate room adjacent to the vessel cells.

The filter-demineralizer system also services the torus water cleanup system for the purification of suppression pool water.

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

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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/ft^2 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.

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The influent and effluent water of the Spent Fuel Pool Demineralizer is continuously monitored by on-line pH and conductivity instrumentation. In addition grab samples of the influent water will be analyzed 1/week for Cl and for gamma isotopic and 1/month for heavy metals, and the effluent water will be analyzed weekly for Cl, SiO_2, suspended solids, H-3 and for gamma isotopic, when the cleanup system is in operation.

Decontamination factors (df) of > 10 are expected for any Cl present and > 5 for isotopes of I and Co. Resin beds will be regenerated and/or replaced when these df's are not achieved.

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The pressure drop across the Demineralizer is continously monitored and when the DP increases to a predetermined level the ion exchange media will be replaced. Typically this level is 30 PSID.

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The Spent Fuel Pool Demineralizer will be operated as required to maintain radiation levels on the refueling platform less than 2 mrem/hr.

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Figure 9.2-13 shows the refueling water transfer pumps. Manual valves are aligned to establish the fill flow path, and the pumps are manually started. Provision is made to permit filling the cask pool or the reactor well independently.

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After refueling or spent fuel shipping activities are completed, either the reactor well and the dryer and separator storage pool, or the cask pool are drained via gravity drain lines to the refueling water transfer pumps' suction header from which the water is pumped through the ~~spare~~ condensate demineralizer and back to the CST. Alternately, the reactor well, dryer and separator storage pool, and cask pool can be drained via gravity drain lines to the fuel pool pumps' suction header from which the water is pumped through the fuel pool filter-demineralizers and back to the CST. During refueling operations, a portion of the cooling system flow is diverted from the fuel pool return line to the reactor well via the reactor well diffusers. The recirculation pattern established by the diverted flow allows the heated water that rises above the reactor core to be cooled in the fuel pool heat exchangers. This supplements the parallel RHR system (operating in the shutdown cooling mode) decay heat removal from the core region. When the shipping cask contains spent fuel and is in the cask pool, a portion of the FPCC system flow is diverted from the fuel pool diffusers to the cask pool via the cask pool diffuser. When the RHR system is operated in parallel with the FPCC system to provide fuel pool cooling during the full core unload case, one RHR pump takes the suction from the skimmer surge tanks, circulates the water through one RHR heat exchanger, and returns it to the spent fuel pool via the two RHR intertie return diffusers.

The cask pool is filled via the refueling fill line and drained through a condensate demineralizer or the fuel pool filter-demineralizers in the same manner as the refueling volume is filled and drained. Filling of the cask pool is normally done prior to spent fuel loading into the cask, and draining is normally accomplished after cask loading.

The FPCC system design heat load is 16.0×10^6 Btu/h. This is the decay heat expected from 16 ~~15~~ consecutive refuelings. ~~, rounded upward to the nearest million Btu/h.~~ The FPCC system's maximum heat load is 34.2 ~~31.0~~ $\times 10^6$ Btu/h. This is the decay heat expected if it becomes necessary to unload the entire core from the reactor and store it in the pool, which already contains spent fuel from thirteen ~~two~~ previous refuelings. For this core unload design condition, an RHR heat exchanger is operated in parallel with the FPCC system. The RHR system is only interconnected when the reactor is shut down, and larger-than-normal batches of spent fuel, such draining the suppression pool if it is ever necessary. In this mode of operation, the torus water cleanup pump takes suction from the torus and circulates the water through a fuel pool filter-demineralizer and to the CST. Operator action is necessary to terminate torus water cleanup operation, except on low pump suction flow.

9.1.3.3 Safety Evaluation

The FPCC system cooling loop (skimmers, skimmer surge tanks, fuel pool cooling pumps, fuel pool heat exchangers, interconnecting loop piping), and the emergency fuel pool water makeup system are designed to the requirements of Seismic Category I, except for the surge tanks. The surge tanks are of non-Seismic Category I design, but are embedded in a Seismic Category I concrete structure that provides the pressure boundary for this part of the FPCC system cooling loop. The interconnecting piping between RHR and the FPCC system is designed to Seismic Category I requirements. insert 1

The cooling water return lines to the spent fuel pool, associated with both the FPCC and the RHR systems, penetrate the walls of the spent fuel pool horizontally above the normal pool water level. Each of these cooling water return lines is provided with two vacuum breakers to prevent the water from being siphoned out of the pool. No piping connections are made to the pool below the normal water level to prevent any accidental lowering of the water level. Therefore, there is no operator error or FPCC system malfunction that could result in draining the spent fuel pool and uncovering the stored spent fuel. The fuel pool structures are also designed to Seismic Category I requirements. If a line break occurs in the non-Seismic Category I purification loop, the remotely operated purification loop isolation valves close automatically on surge tank low-low level or by operator action.

Any leakage between the fuel pool gates, cask pool gates, or through the vessel to drywell seal or drywell to reactor well seal is alarmed in the MCR. A segmented leak channel system behind the liner weld seams is provided to detect fuel pool, cask pool, reactor well, and dryer and separator pool leakage.

The torus water cleanup system suction and return piping from the torus, out through and including the primary containment isolation valves on each line, is designed to Seismic Category I requirements. The torus water cleanup system piping to and from

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during construction

The surge tanks were designed to withstand an external loading of 690 lb/ft^2 during construction. The actual concrete loading (approximately 300 lb/ft^2) was lower than the design value due to the use of a slower pour rate. This external loading induced a stress level less than one half of the design stress level in the tank shell.

An analysis has been performed to determine the effect of seismic loads on the skimmer surge tanks. This analysis indicates that the induced stresses resulting from the seismic loads are insignificant (approximately 1% of the stresses due to concrete placement) and that the skimmer surge tanks will not fail ~~in such a manner as to plug the tank drains.~~ following a safe shutdown earthquake.

The combined use of the techniques mentioned allows an accurate assessment of the SFPFD and permits the determination of when a specific unit should be changed.

Reactor well water level is monitored in the CRIDS, and an annunciator alarm is provided in the MCR to indicate a low reactor well water level during refueling. An interlock trips the refueling water transfer pumps on low reactor well level when the well is draining back to the CST after fuel transfer.

The torus water cleanup pump is started and stopped from the FPCC filter-demineralizer panel and the pump is stopped automatically by low suction flow. Low suction flow is alarmed on the FPCC filter-demineralizer panel. A pressure indicator is located in the pump discharge line.

9.1.3.6 SRP Rule Review

Acceptance Criterion II.1.d.(4) of SRP 9.1.3 limits the water temperature in the fuel pool to 140°F at the maximum heat load with the normal cooling system operating in a single active failure condition.

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The bulk water temperature in the fuel pool ~~may exceed 140°F with only one heat exchanger in service after the first refueling cycle.~~ However, the RHR system can be manually aligned to provide supplemental cooling in order to avoid bulk water temperatures in excess of 140°F.

9.1.4 FUEL HANDLING SYSTEM

9.1.4.1 Design Bases

The fuel handling system is designed to provide a safe and effective means for transporting and handling fuel from the time it reaches the plant until the time it leaves the plant after post-irradiation cooling. Safe handling of fuel includes design considerations for maintaining occupational radiation exposures as low as reasonably achievable (ALARA) during transportation and handling.

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... could reach 152°F for the worst case single active failure of one FPCC pump with the maximum normal heat load of 16.1 ~~16.0~~ $\times 10^6$ Btu/hr. The radiological consequences of the fuel pool temperature reaching 152°F have been evaluated. The resultant doses will not exceed 10CFR20 limits at the site boundary.

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TABLE 9.1-1

FUEL POOL COOLING AND CLEANUP SYSTEM AND TORUS WATER CLEANUP SYSTEM DESIGN PARAMETERS

Skimmer Surge Tanks	
Type	Vertical, cylindrical
Quantity	2
Design pressure, psig	0
Design temperature, °F	212
Capacity, gallons	3750
Fuel Pool Cooling and Cleanup System Pumps	
Type	Horizontal, centrifugal, single-stage
Quantity	2
Design pressure, psig	150
Design temperature, °F	212
Rated flow per pump, gpm	700
Developed head (TDH) at rated flow, feet	257
Motor horsepower, each	75
Fuel Pool Heat Exchangers	
Type	Plate
Quantity	2
Design pressure, psig	
Cold side	150
Hot side	175
Design temperature, °F	
Cold side	150
Hot side	212
Rating, Btu/h (at ~~95~~ 95°F SACS and 135°F Fuel Pool)	8.2 ~~6.0~~ x 10^6
Flow, each, gpm	700

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TABLE 9.1-2

FUEL POOL COOLING AND CLEANUP SYSTEM HEAT REMOVAL CAPACITY AND MAKEUP REQUIREMENTS

Parameter	Value at Normal Heat Load	Value at Maximum Heat Load
Quantity of fuel	1/3 of core: 4-1/2 yr irradiation time 8 days decay time	1/3 of core: 4-1/2 yr irradiation time 10 days decay time
	1/3 of core: 4-1/2 yr irradiation time 556 days decay time	1/3 of core: 3 yr irradiation time 10 days decay time
	1/3 of core: 4-1/2 yr irradiation time 1104 days decay time	1/3 of core: 1-1/2 yr irradiation time 10 days decay time
	1/3 of core: 4-1/2 yr irradiation time 1652 days decay time	1/3 of core: 4-1/2 yr irradiation time 558 days decay time
	1/3 of core: 4-1/2 yr irradiation time 2200 days decay time	1/3 of core: 4-1/2 yr irradiation time 1105 days decay time
	1/3 of core: 4-1/2 yr irradiation time 2748 days decay time	1/3 of core: 4-1/2 yr irradiation time 1652 days decay time
	1/3 of core: 4-1/2 yr irradiation time 3296 days decay time	1/3 of core: 4-1/2 yr irradiation time 2200 days decay time
	1/3 of core: 4-1/2 yr irradiation time 3844 days decay time	1/3 of core: 4-1/2 yr irradiation time 2748 days decay time
	1/3 of core: 4-1/2 yr irradiation time 4392 days decay time	1/3 of core: 4-1/2 yr irradiation time 3296 days decay time
	1/3 of core: 4-1/2 yr irradiation time 4940 days decay time	1/3 of core: 4-1/2 yr irradiation time 3844 days decay time
	1/3 of core: 4-1/2 yr irradiation time 5488 days decay time	1/3 of core: 4-1/2 yr irradiation time 4392 days decay time
	1/3 of core: 4-1/2 yr irradiation time 6036 days decay time	1/3 of core: 4-1/2 yr irradiation time 4940 days decay time
	1/3 of core: 4-1/2 yr irradiation time 6584 days decay time	1/3 of core: 4-1/2 yr irradiation time 5488 days decay time
	1/3 of core: 4-1/2 yr irradiation time 7132 days decay time	1/3 of core: 4-1/2 yr irradiation time 6036 days decay time
	1/3 of core: 4-1/2 yr irradiation time 7680 days decay time	1/3 of core: 4-1/2 yr irradiation time 6584 days decay time
	1/3 of core: 4-1/2 yr irradiation time 8228 days decay time	1/3 of core: 4-1/2 yr irradiation time 7132 days decay time
		1/3 of core: 4-1/2 yr irradiation time 7680 days decay time
		1/3 of core: 4-1/2 yr irradiation time 8228 decay time
Normal design heat load	16.1 ~~10~~ x 10^6 Btu/h	
Number of pumps required	2	
Number of heat exchangers required	2	RHR system
Maximum design heat load		~~[illegible]~~ 54.2 x 10^6 Btu/h
Water makeup requirements due to evaporation losses:		
Makeup during normal operation	2 gpm	
Makeup rate for refueling	5 gpm	

TABLE 9.1-18

DECAY HEAT AND EVAPORATION RATES FOR LOSS OF SPENT FUEL POOL COOLING

Description of the event		Normal heat load in the spent fuel pool (16.1 x 10^6 BTU/hr)	Maximum heat load in the spent fuel pool (~~31.[illegible]~~ 34.2 x 10^6 BTU/hr) (4)
A	Time to reach 212°F	17.2 hrs(5)	~~8.9~~ 8.03 hrs
B	Evaporation rate	34.4 gpm	~~66.7~~ 73.5 gpm
C	Time required to initiate makeup water	2 hrs(1) 1/2 hr(2) 20 hrs(3)	~~1/2 hr(4)~~ 2 hrs (1) 1/2 hr (2)

Notes:

(1) An estimated time of 2 hrs would be required to couple the fire hose fill connections to the Seismic Category I SSWS loops to provide fresh water makeup to the fuel pool.

(2) It has been conservatively estimated that the SSWS can be initiated within 1/2 hr by operator action in the MCR to provide makeup to the fuel pool.

(3) It has been conservatively estimated that after 20 hrs one RHR pump loop and the associated heat exchanger can be used for fuel pool cooling. ~~RHR cooling can be initiated from the MCR.~~

(4) Since the entire core is in the fuel pool, the RHR system can be made availabe ~~within approximately 1/2 hr~~ for fuel pool cooling. ~~by operator action in the MCR.~~

(5) This assumes a normal maximum heat load after ~~15~~ 16 consecutive refuelings.

QUESTION 410.46 (SECTION 9.1.3)

Verify that the normal heat load after refueling can be removed by using one spent fuel pool cooling system pump and both heat exchangers. With this system configuration, verify that the pool water temperature will remain less than 140°F and specify the length of time that that (SIC) second heat exchanger is required. If this cannot be verified, justify this deviation from the Standard Review Plan.

RESPONSE

The fuel pool temperature could ~~exceed 140°F~~ reach 152°F with normal maximum heat load in the fuel pool, one spent fuel pool cooling pump and both heat exchangers operating for fuel pool cooling. Insert C

With the above system configuration it has been conservatively estimated that after 90 days the fuel pool heat load will be such that only one fuel pool heat exchanger is required for fuel pool cooling.

Insert C.

section 9.1.3.6 has been revised to reflect the above and to address the radiological consequences of a pool temperature of 152°F.

QUESTION 410.47 (SECTION 9.1.3)

Verify that the decay heat loads are based on NUREG-0800, Standard Review Plan, Section 9.1.3 and Branch Technical Position ASB 9-2.

RESPONSE

The fuel pool heat loads are calculated based on NUREG-0800, Standard Review Plan, Section 9.1.3 and Branch Technical Position ASB 9-2, except for the following:

1. For HCGS "annual refueling" means 18 month refueling.
2. The decay time is assumed to be 8 days for calculating the normal heat load and 10 days for the maximum heat load. *

Table 9.1-2 describes the basis for calculating the normal and maximum heat loads.

* These times are somewhat longer than recommended in SRP 9.1.3, but are consistent with the times presented in "BWR Servicing and Refueling Improvement Program - Phase I Summary Report" (NEDG 21860, Sept., 1978).

QUESTION 410.55 (SECTION 9.1.3)

Provide a discussion of the means to provide cooling to the spent fuel pool after a safe shutdown earthquake which fails the skimmer surge tanks and plugs the tank drains. The results will be the loss of the spent fuel pool cooling system pumps due to cavitation from an isolated suction line, loss of offsite power from the earthquake, and the unavailability of the RHR system from the loss of the common suction with the spent fuel pool cooling pumps. The worst single active failure should be considered as part of the discussion. If the pool is allowed to boil, then consideration must be given to the time required to clear the skimmer tank drains as compared to the minimum time required to achieve boiling; the continued reduction in worker efficiency as the ambient air temperature, humidity, and radioactivity increases; and the time required to bring the reactor to cold shutdown and thereby have an RHR cooling loop available to cool the pool.

RESPONSE

Consideration of multiple failures of non-Seismic Category I components following a safe shutdown earthquake is beyond the design basis for HCGS. In particular, the postulated failure of both skimmer surge tanks is not considered credible because these "tanks" are, in fact, steel-lined voids in the Seismic Category I spent fuel pool wall.

Section 9.1.3.2 discusses the backup sources of makeup water available to supply the pool in the event normal cooling is lost and RHR cooling is not available.

section 9.1.3.3 has been revised to respond to this question.

Rev 1

DSER Open Item No. 151 (DSER Section 9.4.1)

CONTROL STRUCTURE VENTILATION SYSTEM

The CRS and CREF systems take outside air from a common tornado-missile-protected air intake. The air intake for the CERS system is also tornado missile protected; however, there is no protection for the nonsafety-related WAS system intake. The exhaust for the CABE, WAE, CASE, and CAE systems are tornado missile protected. Thus, the staff concludes that the requirements of GDC 4, "Environmental and Missile Design Bases," are satisfied. The air intakes have no chlorine monitoring capability but do have radiation monitoring capability. Signals from the radiation detectors alarm in the control room, automatically isolate the fresh air intake from the control room HVAC system, and automatically start the CREF system to purify the fresh air. There is no automatic operation associated with the redundant CREF system train upon loss of the operating system. The CRS and CREF systems are designed to maintain the operability of the equipment in the control room. The control room systems are designed to maintain the control room under a positive pressure to minimize infiltration of gases into the control room except during 100% recirculation operation. Thus, the staff concludes that the requirements of GDC 19, "Control Room," and the guidelines of Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," Positions C.3, C.7, and C.14, are satisfied. We cannot conclude that the guidelines of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," Positions C.4a and C.4d are satisfied.

The CRS, CREF, and CERS systems consist of two 100% capacity trains of filters. The CREF system consists of a prefilter, a HEPA filter, a charcoal filter, and a fan in series for the removal of radioactivity. The CRS and CERS systems consist of a prefilter, high efficiency filter, and a fan. There is no filtration of the exhaust; however, it is isolated upon a high radiation signal.

Chilled water is supplied to the two 50% capacity cooling coils in each of the air handler units. The maximum ambient temperature for which one train will maintain the proper environment is 94°F. The applicant must demonstrate that one train of ventilation systems can maintain the compartment environmental conditions within the qualification limits with an outside ambient temperature of 102°F for all design basis accidents with the loss of the redundant ventilation systems. Based on the above, we cannot conclude that the requirements of General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment," and the

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guidelines of Regulatory Guides 1.52, "Design, Testing, and Maintenance Criteria for Atmospheric Cleanup System Air Filtration and Adsorption Units of Light Water-Cooled Nuclear Power Plants," Position C.2, and 1.140, "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light Water-Cooled Nuclear Power Plants," Positions C.1 and C.2, are satisfied with respect to ensuring environmental limits for proper operation of plant controls under all normal and accident conditions, including LOCA conditions.

Based on the above, the staff concludes that the CSV systems are in conformance with the requirements of the GDC 2, 4, and 19 with respect to protection against natural phenomena, tornado missile protection, and control room environmental conditions and the guidelines of RGs 1.29, Positions C.1 and C.2, and 1.78, Positions C.3, C.7, and C.14, relating to the seismic classification and protection against hazardous chemical release and is, therefore, acceptable. We cannot conclude that the CSV systems are in conformance with the requirements of General Design Criterion 60 with respect to control of radioactive releases and the guidelines of Regulatory Guide 1.52, Position C.2, 1.95, Positions C.4.a and C.4.d, and 1.140, Positions C.1 and C.2, relating to the design for emergency operation, protection of personnel against a chlorine gas release, and normal operation. We will report resolution of this item in a supplement to this SER. The HVAC systems which make up the CSV systems do not meet the acceptance criteria of SRP Section 9.4.1.

RESPONSE

Evaluation of accidents relating to the release of toxic chemicals including chlorine is addressed in FSAR Section 2.2.3.1.3.

Also, per DSER Section 6.4, Page 6-3:

"With respect to toxic gas protection, the staff's evaluation in accordance with SRP Section 6.4, RGs 1.78 and 1.95 indicated that there is no danger to control room personnel from toxic chemicals, including chlorine, stored onsite or offsite, or transported nearby (See Section 2.2.3)."

Section 9.4.1.3 has been revised to include reference to Section 2.2.3.1.3.

The CRS system provides cooling (with chilled water cooling coils) during normal operating conditions. The system also provides cooling, in conjunction with the CREF unit, in the event of an accident condition.

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The function is either:

1. 1000 cfm outside ~~our~~ air makeup mixed with 3000 cfm of room return air diverted through the CREF unit. The balance of air is recirculated from the air conditioned space or,

2. A 100% recirculation mode, i.e., without outside air and with the use of the CREF unit.

See FSAR Section 9.4.1.2.3.

Function Mode ~~2~~ 1 is selected in the event of an accident condition. When the outside ambient temperature condition is 102°F, 1000 cfm air is a minimal quantity (~~A~~pproximately 5.4% of the total air supply) which will increase the suppl~~ies~~y air temperature by less than 1°F. Therefore, this increase in temperature, resulting in a control room temperature of 77°F persisting for a total of 180 hours (i.e, 6 hours per day for 30 days) will not affect the operation of the plant controls due to the use of cooling coils as stated above. Since neither outside air is brought into the system nor is the control room exposed to solar load, outside ambient temperature of 102°F has no effect on Function Mode 2.

Refer to the following sections for further safety considerations included in the design of the safety-related control area HVAC systems:

a. Protection from wind and tornado effects - Section 3.3

b. Flood design - Section 3.4

c. Missile protection - Section 3.5

d. Protection against dynamic effects associated with the postulated rupture of piping - Section 3.6

e. Environmental design - Section 3.11

f. Fire protection - Section 9.5.1.

g. Toxic chemicals - Section 2.2.3.1.3

9.4.1.4 <u>Tests and Inspections</u>

The CRS, CERS, CREF, and CABE systems and their components are tested in a program consisting of the following:

a. Factory and in-situ qualification tests (see Table 9.4-6)

b. Onsite preoperational testing (see Chapter 14)

c. Onsite operational periodic testing (see Chapter 16).

Written test procedures establish minimum acceptable values for all tests. Test results are recorded as a matter of performance record, thus enabling early detection of faulty operating performance.

All equipment is factory inspected and tested in accordance with the applicable equipment specifications, codes, and quality assurance requirements. Refer to Table 9.4-6 for details of inspection and testing.

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ENGINEERED SAFETY FEATURES VENTILATION SYSTEM

The safety related systems are designed to Seismic Category I, Quality Group C requirements and are housed in the seismic Category I, flood and tornado protected auxiliary building, thereby satisfying the requirements of GDC 2 and the guidelines of RG 1.29, Positions C.1 and C.2. The applicant has provided tornado missile protection for the inlet and outlet louvers. The system is separated from high-energy piping systems and internally generated missiles. [The applicant has not specified the maximum temperature inside the structure with all equipment running during a 102°F summer day. The 102°F day is the maximum summer temperature recorded between 1948 and 1981 (refer to FSAR Table 2.3-13). Therefore, we cannot conclude that the requirements of General Design Criterion 4 are satisfied.] The inlet louvers have tornado-missle-protected barriers and are more than 30 ft above plant grade; thus, the staff concludes that the guidance of Item 2, Subsection A, of NUREG-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability," and therefore, the pertinent requirements of GDC 17, "Electric Power System," relating to the protection of essential electrical components from failure due to the accumulation of dust and particulate material, are satisfied.

RESPONSE

A. Service Water Intake Structure (SWIS)

With an extreme outdoor air temperature of 102°F, the SWIS room ambient temperature will rise (from 104°F with design outdoor air temperature of 94°F) to approximately 113°F. The manufacturer's design information and/or the equipment environmental qualification reports for all active, safety-related equipment and instrumentation in the service water intake structure which could be affected by temperature has been reviewed. A temperature of 113°F will not cause the failure of any of this equipment or instrumentation. This temperature persisting for a total of 180 hours (i.e., 6 hours per day for 30 days) will not have a signifcant impact on the life of this equipment or instrumentation.

B. Standby Diesel Generator (SDG) Area

Section 9.4.6.1 has been revised to indicate maximum space design temperatures. An extreme outdoor air temperature of

102°F would have little or no effect on SDG area HVAC systems or safety-related equipment. Individual HVAC systems within the SDG area are discussed below:

1. <u>IE Panel Room Supply</u>

   The IE panel room supply unit mixes 7000 cfm outside air with 34000 cfm return air and further cools this mixture using cooling coils before it is distributed. A rise in outdoor temperature from 94°F to 102°F would result in less than a 1.5°F rise in the mixed air temperature entering the cooling coil. Because of reserve capacity in the cooling coil, space temperatures will rise less than 1.5°F.

2. <u>SDG Air Recirculation</u>

   The SDG air recirculation system recirculates 100 percent room air and is designed to maintain a space maximum of 120°F; thus, the system would be unaffected by a rise in outside air temperature to 102°F.

3. <u>Switchgear Room Cooling</u>

   The switchgear room cooling units each mix 1840 cfm outside air with 9360 cfm return air and further cool this mixture using cooling coils before it is distributed. A rise in outdoor temperature from 94°F to 102°F would result in less than a 1.5°F rise in the mixed air temperature entering the cooling coil. Because of reserve capacity in the cooling coil, space temperatures will rise less than 1.5°F.

4. <u>Safety-Related Battery Room Exhaust</u>

   Air is supplied to safety-related battery rooms by either the IE panel room supply system or the switchgear room cooling system and is then exhausted by this system. Based on discussions above, the temperature in the safety-related battery rooms will rise no more than 1.5°F above the design maximum temperature.

   Temperature increases of less than 1.5°F ~~for short periods~~ would have no effect on safety-related equipment operation or environmental qualification.

   Refer to DSER Open Item No. 1.

   persisting for a total of 180 hours (i.e, 6 hours per day for 30 days) indication for these locations is provided in the main control room.

9.4.5.6 SRP Rule Review

Justifications for deviations with SRP Section 9.4.5, Engineered Safety Feature (ESF) Ventilation Systems, are presented in SRP Sections 9.4.1 and 9.4.2, which address the specific ESF ventilation systems of HCGS.

9.4.6 STANDBY DIESEL GENERATOR AREA VENTILATION SYSTEMS

9.4.6.1 Design Bases

The standby diesel generator (SDG) area ventilation systems maintain a suitable operating environment for the SDG rooms, the safety- and nonsafety-related battery rooms, switchgear rooms, SDG fuel oil storage rooms, electrical chases, corridors in the diesel area, and the SDG Class 1E panel room during all modes of plant operation. The heating, cooling, and ventilating systems for the SDG area consist of both safety-related and nonsafety-related systems. The seismic classification and corresponding codes and standards that apply to the design of the system are discussed in Section 3.2.

Insert A

9.4.6.2 System Description

The SDG area is provided with the separate ventilation systems listed below and shown on Figures 9.4-15 and 9.4-16. Equipment design parameters are listed in Table 9.4-16. The systems are:

a. Diesel area supply system - This system is nonsafety-related. It is composed of two 50%-capacity heating and ventilating units. It supplies air to the SDG area corridor, stairwells, and the electrical chases. Outside air is taken from a Seismic Category I plenum and passed through an automatic outside air intake damper, low efficiency and high efficiency filters, an electric heating coil, a centrifugal supply fan provided with automatic inlet vanes, and an automatic supply air shutoff damper.

INSERT A

INSERT TO FSAR SEC 9.4.6.1

The heating, cooling and ventilation systems for the stand-by diesel generator area is designed to maintain the following space temperatures during normal plant operations, based on outside design temperature of 94°F dry bulb / 78°F wet bulb:

a. 104°F maximum in the H & V equipment rooms, corridors, electrical chases, diesel fuel tank rooms and the diesel generator recirculation fan rooms.

b. 120°F maximum in the diesel generator rooms when the diesel generators are energized.

c. 85°F maximum in the control equipment rooms, inverter rooms, battery charger rooms and the diesel generator control rooms.

d. 90°F maximum in the switchgear rooms.

e. 77°F ± 3 F in the battery rooms.

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The discharge shutoff damper automatically opens when the fan starts. The fans are started by handswitches located on the local panel.

The two safety-related battery rooms at elevation 163 feet 6 inches are provided with two 100%-capacity exhaust fans. Makeup air to these battery rooms is provided by the diesel area Class 1E panel room supply system. Each fan is provided with a manual inlet shutoff damper and an automatic discharge shutoff damper, and a tornado protection damper. During LOP, the fans are automatically connected to emergency Class 1E power from the SDG. The automatic discharge shutoff damper opens when the fan starts. A low flow computer input actuates an alarm in the main control room upon loss of airflow, and starts the redundant fan automatically. The fans are started by handswitches located on the local panel.

d. Diesel area nonsafety-related battery room exhaust system - The two nonsafety-related battery rooms at elevation 163 feet 6 inches are provided with two 100%-capacity exhaust fans. Each fan has a manual inlet shutoff damper, and an automatic discharge shutoff damper, and a tornado protection damper. Makeup air to these battery rooms is provided by the diesel area Class 1E panel room supply system. During LOP, the fans can be manually connected to SDG-backed non-Class 1E power from the SDG. A low flow computer input actuates an alarm in the main control room upon loss of airflow, and automatically starts the redundant fan. The automatic discharge damper opens when the fan starts. The fans are started by handswitches located on the local panel.

e. Switchgear room cooling systems - These are safety-related systems. Each of the four switchgear rooms is provided with one Seismic Category I, full-capacity air cooling unit that has a centrifugal supply fan, a tornado protection check damper at its outside air intake duct ~~louver~~, a low efficiency filter, and two 100%-capacity chilled water cooling coils. The air cooling unit can be isolated by the automatic outside air shutoff damper and by manual dampers located in the discharge and return ducts. A mixture of outside air and return air enters the switchgear room unit cooler for processing. The conditioned air is supplied to the switchgear room, battery charger room, battery room, and SDG control room of each respective SDG. Cooling coils are supplied with chilled water from the safety-related control area chilled water system. Chilled water piping is arranged so that one coil in each unit receives chilled water from loop A, and the other coil receives chilled water from loop B. During LOP, the cooling units are automatically connected to Class 1E power from the respective SDG that they serve. Each unit cooler can be started by a handswitch located at the local panel.

The low-flow switch for each fan actuates an alarm at the local panel, and in the main control room upon loss of airflow, and stops the operating fan. Alarms are also provided for high-pressure differential across the filter and for high or low return air temperature.

f. Diesel area Class 1E panel room supply system - This system is safety-related and supplies conditioned air to the four battery rooms, ~~eight~~ nine inverter rooms, and two heating, ventilating and air conditioning (HVAC) rooms at elevation 163 feet, and the elevator machine room at elevation 178 feet. It is composed of two 100%-capacity HVAC units. One unit runs while the other is on standby. The standby unit will automatically start upon failure of the operating unit. Outside air for each unit is taken from a separate Seismic Category I plenum. Each unit has a low and a high efficiency filter, an electric heating coil, a chilled water coil, and a centrifugal supply fan provided with automatic inlet vanes. The outside air return duct and discharge air ducts are provided with automatic shutoff dampers. The outside air duct is also provided with a tornado protection check damper. A flow controller is provided that ensures a constant air volume. The cooling coil is supplied with chilled water from the auxiliary building control area chilled water system. Water piping is arranged so that the coil of one unit receives chilled water from loop A and the coil of the other unit receives chilled water from loop B. During LOP, the units are automatically connected to emergency Class 1E power from the SDG. Each unit cooler can be started by a handswitch located at the local panel.

The low-flow switch actuates a local alarm upon loss of airflow and starts the standby units. Local alarms are also provided for high-pressure differential across the

DSER Open Item No. 244 (DSER Section 8.3.3.3.1)

ANALYSIS AND TEST TO DEMONSTRATE ADEQUACY OF LESS THAN SPECIFIED SEPARATION

The applicant, by Amendment 4 to the FSAR, provided a description of physical separation between redundant enclosed raceways (covered trays and open top raceways, and between non-Class 1E trays and Class 1E conduit, as follows:

1. In the cable spreading rooms, the main control room, relay room, and control equipment room, the separation is twelve inches (12") horizontal, and eighteen inches (18") vertical.

2. In all other plant areas, the separation is three feet horizontal and five feet vertical.

The applicant further stated that where the separation distances specified above can not be maintained, cable trays shall either be covered with metal tray covers or an analysis, based on test results, will be performed.

The staff concludes that the above separation meets the guidelines of Regulatory Guide 1.75 and is acceptable except for the following:

(1) The use of 18 versus 36 inches of separation between raceways is evaluated in Section 8.3.3.3.2 of this report, and

(2) The use of an analysis to justify less than specified separation will be pursued with the applicant.

RESPONSE

The response to Question 430.52 has been revised to provide the requested analysis. One copy of each of the following reports are ~~being~~ attached for your use:

1) Wyle Laboraties, Test Report No. 56719, Dated November 20, 1980, prepared for Susquehanna Steam Electric Station for electrical wire and cable isolation barrier ~~test~~ materials test.

2) Franklin Institute Research Laboratories, ~~for~~ Dated March 30, 1977, prepared for Toledo Edison Company for Conduit Separation test Program.

F70(8)