PSEG Public Service Electric and Gas Company

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Robert L. Mittl General Manager
Nuclear Assurance and Regulation

July 27, 1984

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

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

Gentlemen:

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

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

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

In addition, enclosed for your review and approval (see Attachment 4) are the resolutions to those Draft SER open items listed in Attachment 3. A signed original of the required affidavit is provided to document the submittal of these DSER open item responses.

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

Very truly yours,

RL Mittl / RPDouglas

Attachments

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1/1

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.

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 27th day
of July 1984.

David K. Burd

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

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DATE: 7/27/84

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	Open	
2a	2.3.3	Accuracies of meteorological measurements	Complete	7/27/84
2b	2.3.3	Accuracies of meteorological measurements	Complete	7/27/84
2c	2.3.3	Accuracies of meteorological measurements	Complete	7/27/84
2d	2.3.3	Accuracies of meteorological measurements	Open	
3a	2.3.3	Upgrading of onsite meteorological measurements program (III.A.2)	Complete	7/27/84
3b	2.3.3	Upgrading of onsite meteorological measurements program (III.A.2)	Complete	7/27/84
3c	2.3.3	Upgrading of onsite meteorological measurements program (III.A.2)	Open	
4	2.4.2.2	Ponding levels	Open	
5a	2.4.5	Wave impact and runup on service Water Intake Structure	Complete	6/1/84
5b	2.4.5	Wave impact and runup on service water intake structure	Open	
5c	2.4.5	Wave impact and runup on service water intake structure		
5d	2.4.5	Wave impact and runup on service water intake structure	Complete	6/1/84
6a	2.4.10	Stability of erosion protection structures	Open	
6b	2.4.10	Stability of erosion protection structures	Open	
6c	2.4.10	Stability of erosion protection structures	Open	

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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	Open	
7b	2.4.11.2	Thermal aspects of ultimate heat sink	Complete	6/1/84
8	2.5.2.2	Choice of maximum earthquake for New England - Piedmont Tectonic Province	Open	
9	2.5.4	Soil damping values	Complete	6/1/84
10	2.5.4	Foundation level response spectra	Complete	6/1/84
11	2.5.4	Soil shear moduli variation	Complete	6/1/84
12	2.5.4	Combination of soil layer properties	Complete	6/1/84
13	2.5.4	Lab test shear moduli values	Complete	6/1/84
14	2.5.4	Liquefaction analysis of river bottom sands	Complete	6/1/84
15	2.5.4	Tabulations of shear moduli	Complete	6/1/84
16	2.5.4	Drying and wetting effect on Vincentown	Complete	6/1/84
17	2.5.4	Power block settlement monitoring	Complete	6/1/84
18	2.5.4	Maximum earth at rest pressure coefficient	Complete	6/1/84
19	2.5.4	Liquefaction analysis for service water piping	Complete	6/1/84
20	2.5.4	Explanation of observed power block settlement	Complete	6/1/84
21	2.5.4	Service water pipe settlement records	Complete	6/1/84
22	2.5.4	Cofferdam stability	Complete	6/1/84
23	2.5.4	Clarification of FSAR Tables 2.5.13 and 2.5.14	Complete	6/1/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
24	2.5.4	Soil depth models for intake structure	Complete	6/1/84
25	2.5.4	Intake structure soil modeling	Open	
26	2.5.4.4	Intake structure sliding stability	Open	
27	2.5.5	Slope stability	Complete	6/1/84
28a	3.4.1	Flood protection	Complete	7/27/84
28b	3.4.1	Flood protection	Complete	7/27/84
28c	3.4.1	Flood protection	Complete	7/27/84
28d	3.4.1	Flood protection	Complete	7/27/84
28e	3.4.1	Flood protection	Complete	7/27/84
28f	3.4.1	Flood protection	Open	
28g	3.4.1	Flood protection	Complete	7/27/84
29	3.5.1.1	Internally generated missiles (outside containment)	Complete	7/18/84
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	Open	
33	3.5.2	Structures, systems, and components to be protected from externally generated missiles	Open	
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

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ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
36	3.6.2	Postulated pipe ruptures	Complete	6/29/84
37	3.6.2	Feedwater isolation check valve operability	Open	
38	3.6.2	Design of pipe rupture restraints	Open	
39	3.7.2.3	SSI analysis results using finite element method and elastic half-space approach for containment structure	Open	
40	3.7.2.3	SSI analysis results using finite element method and elastic half-space approach for intake structure	Open	
41	3.8.2	Steel containment buckling analysis	Complete	6/1/84
42	3.8.2	Steel containment ultimate capacity analysis	Complete	6/1/84
43	3.8.2	SRV/LOCA pool dynamic loads	Complete	6/1/84
44	3.8.3	ACI 349 deviations for internal structures	Complete	6/1/84
45	3.8.4	ACI 349 deviations for Category I structures	Complete	6/1/84
46	3.8.5	ACI 349 deviations for foundations	Complete	6/1/84
47	3.8.6	Base mat response spectra	Complete	6/1/84
48	3.8.6	Rocking time histories	Complete	6/1/84
49	3.8.6	Gross concrete section	Complete	6/1/84
50	3.8.6	Vertical floor flexibility response spectra	Complete	6/1/84
51	3.8.6	Comparison of Bechtel independent verification results with the design-basis results	Open	

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
52	3.8.6	Ductility ratios due to pipe break	Open	
53	3.8.6	Design of seismic Category I tanks	Complete	6/1/84
54	3.8.6	Combination of vertical responses	Complete	6/1/84
55	3.8.6	Torsional stiffness calculation	Complete	6/1/84
56	3.8.6	Drywell stick model development	Complete	6/1/84
57	3.8.6	Rotational time history inputs	Complete	6/1/84
58	3.8.6	"O" reference point for auxiliary building model	Complete	6/1/84
59	3.8.6	Overturning moment of reactor building foundation mat	Complete	6/1/84
60	3.8.6	BSAP element size limitations	Complete	6/1/84
61	3.8.6	Seismic modeling of drywell shield wall	Complete	6/1/84
62	3.8.6	Drywell shield wall boundary conditions	Complete	6/1/84
63	3.8.6	Reactor building dome boundary conditions	Complete	6/1/84
64	3.8.6	SSI analysis 12 Hz cutoff frequency	Complete	6/1/84
65	3.8.6	Intake structure crane heavy load drop	Complete	6/1/84
66	3.8.6	Impedance analysis for the intake structure	Open	
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

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
69	3.8.6	Factors of safety against sliding and overturning of drywell shield wall	Complete	6/1/84
70	3.8.6	Seismic shear force distribution in cylinder wall	Complete	6/1/84
71	3.8.6	Overturning of cylinder wall	Complete	6/1/84
72	3.8.6	Deep beam design of fuel pool walls	Complete	6/1/84
73	3.8.6	ASHSD dome model load inputs	Complete	6/1/84
74	3.8.6	Tornado depressurization	Complete	6/1/84
75	3.8.6	Auxiliary building abnormal pressure	Complete	6/1/84
76	3.8.6	Tangential shear stresses in drywell shield wall and the cylinder wall	Complete	6/1/84
77	3.8.6	Factor of safety against overturning of intake structure	Complete	6/1/84
78	3.8.6	Dead load calculations	Complete	6/1/84
79	3.8.6	Post-modification seismic loads for the torus	Complete	6/1/84
80	3.8.6	Torus fluid-structure interactions	Complete	6/1/84
81	3.8.6	Seismic displacement of torus	Complete	6/1/84
82	3.8.6	Review of seismic Category I tank design	Complete	6/1/84
83	3.8.6	Factors of safety for drywell buckling evaluation	Complete	6/1/84
84	3.8.6	Ultimate capacity of containment (materials)	Complete	6/1/84
85	3.8.6	Load combination consistency	Complete	6/1/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
86	3.9.1	Computer code validation	Open	
87	3.9.1	Information on transients	Open	
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
95	3.9.3.2	Fatigue evaluation on SRV piping and LCCA downcomers	Complete	6/15/84
96	3.9.3.3	IE Information Notice 83-80	Complete	6/15/84
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

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
100b	3.9.6	10CFR50.55a paragraph (g)	Open	
101	3.9.6	PSI and ISI programs for pumps and valves	Open	
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	Open	
103a2	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Open	
103a3	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Open	
103a4	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Open	
103a5	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Open	
103a6	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Open	
103a7	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Open	
103b1	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Open	
103b2	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Open	
103b3	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Open	
103b4	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Open	
103b5	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Open	

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
103b6	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Open	
103c1	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Open	
103c2	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Open	
103c3	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Open	
103c4	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Open	
104	3.11	Environmental qualification of mechanical and electrical equipment	NRC Action	
105	4.2	Plant-specific mechanical fracturing analysis	Complete	7/18/84
106	4.2	Applicability of seismic andd LOCA loading evaluation	Complete	7/18/84
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	Open	
109b	4.4.7	TMI-2 Item II.F.2	Open	
110a	4.6	Functional design of reactivity control systems	Complete	7/27/84
110b	4.6	Functional design of reactivity control systems	Complete	7/27/84
111a	5.2.4.3	Preservice inspection program (components within reactor pressure boundary)	Complete	6/29/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
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	7/27/84
112b	5.2.5	Reactor coolant pressure boundary leakage detection	Complete	7/27/84
112c	5.2.5	Reactor coolant pressure boundary leakage detection	Complete	7/27/84
112d	5.2.5	Reactor coolant pressure boundary leakage detection	Complete	7/27/84
112e	5.2.5	Reactor coolant pressure boundary leakage detection	Complete	7/27/84
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	Open	
118	5.3.4	Lead factors and neutron fluence for surveillance capsules	Open	

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
119	6.2	TMI item II.E.4.1	Complete	6/29/84
120a	6.2	TMI Item II.E.4.2	Open	
120b	6.2	TMI Item II.E.4.2	Open	
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
124a	6.2.1.5.1	RPV shield annulus analysis	Complete	6/1/84
124b	6.2.1.5.1	RPV shield annulus analysis	Complete	6/1/84
124c	6.2.1.5.1	RPV shield annulus analysis	Complete	6/1/84
125	6.2.1.5.2	Design drywell head differential pressure	Complete	6/15/84
126a	6.2.1.6	Redundant position indicators for vacuum breakers (and control room alarms)	Open	
126b	6.2.1.6	Redundant position indicators for vacuum breakers (and control room alarms)	Open	
127	6.2.1.6	Operability testing of vacuum breakers	Complete	7/18/84
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

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
132	6.2.4	Containment isolation review	Complete	6/15/84
133a	6.2.4.1	Containment purge system	Open	
133b	6.2.4.1	Containment purge system	Open	
133c	6.2.4.1	Containment purge system	Open	
134	6.2.6	Containment leakage testing	Complete	6/15/84
135	6.3.3	LPCS and LPCI injection valve interlocks	Open	
136	6.3.5	Plant-specific LOCA (see Section 15.9.13)	Complete	7/18/84
137a	6.4	Control room habitability	Open	
137b	6.4	Control room habitability	Open	
137c	6.4	Control room habitability	Open	
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	7/27/84
140b	9.1.2	Spent fuel pool storage	Complete	7/27/84
140c	9.1.2	Spent fuel pool storage	Complete	7/27/84
140d	9.1.2	Spent fuel pool storage	Complete	7/27/84
141a	9.1.3	Spent fuel cooling and cleanup system	Open	
141b	9.1.3	Spent fuel cooling and cleanup system	Open	
141c	9.1.3	Spent fuel pool cooling and cleanup system	Complete	6/29/84

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	Open	
141e	9.1.3	Spent fuel pool cooling and cleanup system	Open	
141f	9.1.3	Spent fuel pool cooling and cleanup system	Open	
141g	9.1.3	Spent fuel pool cooling and cleanup system	Open	
142a	9.1.4	Light load handling system (related to refueling)	Closed (5/30/84-Aux.Sys.Mtg.)	6/29/84
142b	9.1.4	Light load handling system (related to refueling)	Closed (5/30/84-Aux.Sys.Mtg.)	6/29/84
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	Open	
144b	9.2.1	Station service water system	Open	
144c	9.2.1	Station service water system	Open	
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
147a	9.3.1	Compressed air systems	Open	
147b	9.3.1	Compressed air systems	Open	

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
147c	9.3.1	Compressed air systems	Open	
147d	9.3.1	Compressed air systems	Open	
148	9.3.2	Post-accident sampling system (II.B.3)	Open	
149a	9.3.3	Equipment and floor dr[illegible]age 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	Open	
151a	9.4.1	Control structure ventilation system	Complete	7/27/84
151b	9.4.1	Control structure ventilation system	Complete	7/27/84
152	9.4.4	Radioactivity monitoring elements	Closed (5/30/84- Aux.Sys.Mtg.)	6/1/84
153	9.4.5	Engineered safety features ventila- tion system	Complete	7/27/84
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	
157	9.5.1.4.e	Cable tray protection	Open	
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	Open	

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
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
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	Open	

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
176b	14.2	Initial plant test program	Open	
176c	14.2	Initial plant test program	Complete	7/27/84
176d	14.2	Initial plant test program	Complete	7/27/84
176e	14.2	Initial plant test program	Complete	7/27/84
176f	14.2	Initial plant test program	Open	
176g	14.2	Initial plant test program	Open	
176h	14.2	Initial plant test program	Open	
176i	14.2	Initial plant test program	Complete	7/27/84
177	15.1.1	Partial feedwater heating	Complete	7/18/84
17[illegible]	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	Open	
184	7.2.2.1.e	Failures in reactor vessel level sensing lines	Complete	7/27/84
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	Open	

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
187	7.2.2.4	Lifting of leads to perform surveillance testing	Open	
188	7.2.2.5	Setpoint methodology	Open	
189	7.2.2.6	Isolation devices	Open	
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	6/1/84
193	7.3.2.1.10	Manual initiation of safety systems	Open	
194	7.3.2.2	Standard review plan deviations	Complete	6/1/84
195a	7.3.2.3	Freeze-protection/water filled instrument and sampling lines and cabinet temperature control	Open	
195b	7.3.2.3	Freeze-protection/water filled instrument and sampling lines and cabinet temperature control	Open	
196	7.3.2.4	Sharing of common instrument taps	Open	
197	7.3.2.5	Microprocessor, multiplexer and computer systems	Complete	6/1/84
198	7.3.2.6	TMI Item II.K.3.18-ADS actuation	Open	
199	7.4.2.1	IE Bulletin 79-27-Loss of non-class IE instrumentation and control power system bus during operation	Open	
200	7.4.2.2	Remote shutdown system	Complete	6/1/84
201	7.4.2.3	RCIC/HPCI interactions	Open	
202	7.5.2.1	Level measurement errors as a result of environmental temperature effects on level instrumentation reference leg	Open	

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
203	7.5.2.2	Regulatory Guide 1.97	Open	
204	7.5.2.3	TMI Item II.F.1 - Accident monitoring	Open	
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	Open	
208	7.7.2.2	Multiple control system failures	Open	
209	7.7.2.3	Credit for non-safety related systems in Chapter 15 of the FSAR	Complete	6/1/84
210	7.7.2.4	Transient analysis recording system	Complete	6/1/84
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
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

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ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
216b	5.3.1	Reactor vessel materials	Complete	7/27/84
217	9.5.1.1	Fire protection organization	Open	
218	9.5.1.1	Fire hazards analysis	Complete	6/1/84
219	9.5.1.2	Fire protection administrative controls	Open	
220	9.5.1.3	Fire brigade and fire brigade training	Open	
221	8.2.2.1	Physical separation of offsite transmission lines	Open	
222	8.2.2.2	Design provisions for re-establishment of an offsite power source	Open	
223	8.2.2.3	Independence of offsite circuits between the switchyard and class IE buses	Open	
224	8.2.2.4	Common failure mode between onsite and offsite power circuits	Open	
225	8.2.3.1	Testability of automatic transfer of power from the normal to preferred power source	Open	
226	8.2.2.5	Grid stability	Open	
227	8.2.2.6	Capacity and capability of offsite circuits	Open	
228	8.3.1.1(1)	Voltage drop during transient conditions	Open	
229	8.3.1.1(2)	Basis for using bus voltage versus actual connected load voltage in the voltage drop analysis	Open	
230	8.3.1.1(3)	Clarification of Table 8.3-11	Open	

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
231	8.3.1.1(4)	Undervoltage trip setpoints	Open	
232	8.3.1.1(5)	Load configuration used for the voltage drop analysis	Open	
233	8.3.3.4.1	Periodic system testing	Open	
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	Open	
235	8.3.1.5	Diesel generators load acceptance test	Open	
236	8.3.1.6	Compliance with position C.6 of RG 1.9	Open	
237	8.3.1.7	Decription of the load sequencer	Open	
238	8.2.2.7	Sequencing of loads on the offsite power system	Open	
239	8.3.1.8	Testing to verify 80% minimum voltage	Open	
240	8.3.1.9	Compliance with BTP-PSB-2	Open	
241	8.3.1.10	Load acceptance test after prolonged no load operation of the diesel generator	Open	
242	8.3.2.1	Compliance with position 1 of Regulatory Guide 1.128	Open	
243	8.3.3.1.3	Protection or qualification of Class 1E equipment from the effects of fire suppression systems	Open	
244	8.3.3.3.1	Analysis and test to demonstrate adequacy of less than specified separation	Open	

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
245	8.3.3.3.2	The use of 18 versus 36 inches of separation between raceways	Open	
246	8.?.3.3.3	Specified separation of raceways by analysis and test	Open	
247	8.3.3.5.1	Capability of penetrations to with-stand long duration short circuits at less than maximum or worst case short circuit	Open	
248	8.3.3.5.2	Separation of penetration primary and backup protections	Open	
249	8.3.3.5.3	The use of bypassed thermal overload protective devices for penetration protections	Open	
250	8.3.3.5.4	Testing of fuses in accordance with R.G. 1.63	Open	
251	8.3.3.5.5	Fault current analysis for all representative penetration circuits	Open	
252	8.3.3.5.6	The use of a single breaker to provide penetration protection	Open	
253	8.3.3.1.4	Commitment to protect all Class 1E equipment from external hazards versus only class 1E equipment in one division	Open	
254	8.3.3.1.5	Protection of class 1E power supplies from failure of unqualified class 1E loads	Open	
255	8.3.2.2	Battery capacity	Open	
256	8.3.2.3	Automatic trip of loads to maintain sufficient battery capacity	Open	

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
257	8.3.2.5	Justification for a 0 to 13 second load cycle	Open	
258	8.3.2.6	Design and qualification of DC system loads to operate between minimum and maximum voltage levels	Open	
259	8.3.3.3.4	Use of an inverter as an isolation device	Open	
260	8.3.3.3.5	Use of a single breaker tripped by a LOCA signal used as an isolation device	Open	
261	8.3.3.3.6	Automatic transfer of loads and interconnection between redundant divisions	Open	
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	
TS-10	6.7	MSIV leakage rate		

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
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.'	Stability analysis submitted before second-cycle operation	Open	

ATTACHMENT 2 DATE: 7/27/84

DRAFT SER SECTIONS AND DATES PROVIDED

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

CT:db

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DATE: July 27, 1984

ATTACHMENT 3

OPEN ITEM	DSER SECTION NUMBER	SUBJECT
2a	2.3.3	Accuracies of meteorological measurements
2b	2.3.3	Accuracies of meteorological measurements
2c	2.3.3	Accuracies of meteorological measurements
3a	2.3.3	Upgrading of onsite meteorological measurements program (III.A.2.)
3b	2.3.3	Upgrading of onsite meteorological measurements program (III.A.2.)
28a	3.4.1	Flood Protection
28b	3.4.1	Flood Protection
28c	3.4.1	Flood Protection
28d	3.4.1	Flood Protection
28e	3.4.1	Flood Protection
28g	3.4.1	Flood Protection
110a	4.6.	Functional design of reactivity control system
110b	4.6	Functional design of reactivity control system
112a	5.2.5	Reactor coolant pressure boundary detection
112b	5.2.5	Reactor coolant pressure boundary detection

OPEN ITEM	DSER SECTION NUMBER	SUBJECT
112c	5.2.5	Reactor coolant pressure boundary detection
112d	5.2.5	Reactor coolant pressure boundary detection
112e	5.2.5	Reactor coolant pressure boundary detection
121	6.2.1.3.3	Use of NURE6-0588
122	6.2.1.3.3	Temperature profile
128	6.2.2	Air ingestion
140a	9.1.2	Spent fuel pool storage
140b	9.1.2	Spent fuel pool storage
140c	9.1.2	Spent fuel pool storage
140d	9.1.2	Spent fuel pool storage
144a	9.2.1	Station service water system
144b	9.2.1	Station service water system
144c	9.2.1	Station service water system
149a	9.3.3	Equipment and floor drainage system
149b	9.3.3	Equipment and floor drainage system
151a	9.4.1	Control structure ventilation system
151b	9.4.1	Control structure ventilation system
176c	14.2	Initial plant test program
176d	14.2	Initial plant test program
176e	14.2	Initial plant test program
176i	14.2	Initial plant test program

OPEN ITEM	DSER SECTION NUMBER	SUBJECT
184	7.2.2.1.e	Failures in reactor vessel level sensing lines
206	7.6.2.1	High pressure/low pressure interlocks
211a	4.5.1	Control rod drive structural materials
211b	4.5.1	Control rod drive structural materials
211c	4.5.1	Control rod drive structural materials
211d	4.5.1	Control rod drive structural materials
211e	4.5.1	Control rod drive structural materials
212	4.5.2	Reactor internals materials
213	5.2.3	Reactor coolant pressure boundary material
214	6.1.1	Engineered safety features material
215	10.3.6.	Main steam and feed water system materials
216a	5.3.1	Reactor vessel materials
216b	5.3.1	Reactor vessel materials

ATTACHMENT 4

DSER Open Item No. 2a (Section 2.3.3)

Accuracies of Meteorological Measurements

The applicant states that the entire onsite meteorological measurements system complies with the accuracy specifications presented in RG 1.23, "Onsite Meteorological Programs." However, the applicant has not provided (as requested in RAI 451.10) estimates of the overall system accuracy for each parameter measured. The types of wind speed and direction sensors and recording equipment identified by the applicant in Table 2.3-29 have been used by other applicants and licensees to meet the accuracy specifications of RG 1.23.

Response

For the information requested above, see the response to DSER Open item 3a and b.

DSER Open Item No. 2c (Section 2.3.3)

The metecrological measurements program, during plant operation, will include those parameters currently measured. Meteorological parameters are to be available for display through the radiation monitoring system central radiation processor (CRP), although the method of display has not been specified. Calculations of atmospheric transport and diffusion are also to be available through the CRP, although the models and/or methodology have not been described.

Response

For the information requested above, see the response to DSER Open item 3a and b.

DSER Open Item No. 2b (Section 2.3.3)

The applicant's method for

determining vertical temperature gradient is uncommon, using a matched pair of thermistors. Additional information is required from the applicant to demonstrate that the accuracies of meteorological measurements comply with the system accuracy specifications presented in RG 1.23.

Response

For the information requested above, see the response to DSER Open item 3a and b.

DSER Open Item No. 3a&b (Section 2.3.3)

Upgrading of Onsite Meteorological Measurements Program (III.A.2)

To address the meteorological requirements for emergency preparedness planning outlined in 10 CFR 50.47 and Appendix E to 10 CFR 50, the applicant will be required to upgrade the operational meteorological measurements program to meet the criteria in NUREG-0654, Appendix 2, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants." The upgrades must be in accordance with the schedule of NUREG-0737, III.A.2, "Clarification of TMI Action Plan Requirements," or its supplements. The incorporation of current meteorological data into a real-time atmospheric dispersion model for dose assessments will also be considered as part of the upgraded capability.

Response

For the information requested above, see revised Question Response 451.6, FSAR Section 2.3.3.3 and Table 2.3-29a, b and c.

QUESTION 451.6

Section 2.3.2 provides comparisons of meteorological data collected at the Hope Creek site with data collected at the National Weather Service station at Wilmington, Delaware to determine the representativeness of "the key meteorological parameters crucial to the safety, operation, and construction of Hope Creek Generating Station." Additional meteorological data have also been collected on Artificial Island since 1969 in support of construction and operation of the Salem Nuclear Power Plant. These data can also be compared to data for Hope Creek if different meteorological measurements programs are in use for each Nuclear Power Plant.

a) Provide comparisons of annual wind direction frequencies at the 33-ft, 150-ft, and 300-ft for both the Salem and Hope Creek facilities for the available period of record. Include the number of valid observations and the total possible observations for each period of record.

b) Provide comparisons of annual atmospheric stability distributions (Pasquill stability classes A-G) based on the measurement of vertical temperature gradient between the 300-ft and 33-ft levels and between the 150-ft and 33-ft levels for both the Salem and Hope Creek facilities for the available period or record. Include the number of valid observations and the total possible observations for each period of record.

RESPONSE

a) Annual wind direction frequencies at the 33 ft, 150 ft, and 300 ft levels observed during June 1969 to May 1971 (SGS preoperational data) are shown in Table 451.6-1. The 150 ft wind distribution was derived from January 1970 to May 1971 data. Annual wind direction distribution for the same three levels for the period January 1977 to December 1981 are presented in Tables 451.6-2, 451.6-3 and 451.6-4, respectively.

INSERT A

COMPARISONS

33 feet

Highest wind direction frequencies from the period 1969 to 1971 (SGS) compare favorably with those from 1977 to 1981 (HCGS). The site has a bimodal distribution. SGS data shows the highest frequency of wind directions are SE-SSE-S and W-WNW-NW. HCGS data shows the same pattern. Frequencies other than these modes are evenly distributed throughout the compass points. For all individual years, the data recovery rates are above 90 percent.

INSERT A

Data collection for the period of 1969 to 1971 was from a tower located 1400 feet north of the Hope Creek Reactor Building at a latitude of 39 degrees, 28 minutes, 13 seconds north, and a longitude of 75 degrees, 32 minutes, 12 seconds west. This tower was originally located to support preoperational data collection for the Salem Stations. The tower was relocated to the existing location to facilitate the construction of the Hope Creek Station and the cooling tower.

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Monthly and annual joint frequency distributions of wind speed and direction, based on atmospheric stability classes, are referenced in Section 2.3.2.1.1. The 5-year data base containing hourly site meteorological data from January 1977 to December 1981 was used as input in the analysis.

2.3.3.3 Operational Data Display

The meteorological parameters required by Regulatory Guide 1.97 will be incorporated in the data base to be included on the control room integrated display system (CRIDs) computer. The display of those parameters will be available as part of the display function along with all other related Regulatory Guide 1.97 variables.

The radiation monitoring system central radiation processor (CRP) computer will provide 15-minute average meteorological monitoring system parameters. The parameters available for display are 33-ft wind speed and wind direction, 150-ft wind speed and wind direction, 300-ft wind speed and wind direction, delta-temperature stability indicators between 300- and 33-ft and 150- and 33-ft, as well as precipitation, barometric pressure, solar radiation, and ambient temperature at 33 ft.

Atmospheric transport and diffusion during normal operation will be calculated by the CRP. A method for determining atmospheric transport and diffusion throughout the plume exposure emergency planning zone during emergency conditions is being developed.

INSERT B

2.3.4 SHORT-TERM DIFFUSION ESTIMATES

2.3.4.1 Objective

The objective is to provide conservative and realistic short-term estimates of relative concentration (X/Q), at both the site boundary and the outer boundary of the low population zone (LPZ) following a hypothetical release of radioactivity from HCGS. The assessment is based on the results of atmospheric diffusion modeling and onsite meteorological data.

A ground-level accidental radionuclide release from HCGS is analyzed at various distances. Conservative and realistic X/Q values at the exclusion area boundary (EAB) are derived for the

INSERT B

The postoperational data collection program will consist of an enhancement to the preoperational program. The enhancement consists of a primary and backup data acquisition system (DAS) and a communication computer. A diagram of the system configuration is provided in Figure 2.3-6. A list of the system hardware components is tabulated on Table 2.3-29a. There are no changes to the meterological tower, sensors, power supply, strip chart recorders, or translator cards. The rain gauge has been changed from a weighing bucket to a tipping bucket which meets the NRC criteria of measuring .01 inches of precipitation. This change has been incorporated in Table 2.3-29.

The primary and backup DAS are configured with identical hardware. Each DAS consists of a Hewlett-Packard 9826a Computer, 3497A Data Acquisition/Control Unit, and a Dames & Moore transient protection system. Each DAS is provides with two communication ports, one as a link to the communications computer, and the other for direct dial-up capability. Each DAS provides for up to seven days of fifteen minute averages. The primary DAS collects data from the meterological parameters listed in Table 2.3-29. The backup DAS collects wind speed and direction from the the three tower elevations and two delta T's, as well as the backup meterological tower. The data acquisition system calculate a sigma theta for each of the three level wind directions.

The communications computer which consists of a DEC PDP 11/23 computer and RX02 dual disk drive. The communications computer is configured with nine I/O ports. The I/O ports support data transfer/interrogation with the Salem Control Room the Hope Creek Radiation Monitoring System via a meteorological system link (which incorporates a HP9915 computer) as well as three dial up ports. The communication computer also supports a display unit in the the Hope Creek EOF as well as communication to the primary and backup DAS.

System accuracy is presented on Tables 2.3-29b and 2.3-29c.

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INSERT B
(CONTINUED)

The postoperational data collection program also includes an additional meterological tower identifies as a backup meterological tower, consisting of a 10 meter telephone poll. The backup tower is located aproximately 500 feet south of the primary meterological monitoring tower. Backup meterological data provides wind speed, wind direction, and a computed sigma theta. Backup meteorological data provides wind speed and wind direction and a computer sigma theta.

The CRP display of meteorological parameters will be provided by "menu" driven access.

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TABLE 2.3-25 (cont)

Height Above Tower Base, ft	Sensed Parameter	Recorded Parameter	Instrument and Characteristic	Strip Recorders
33	Wind speed	Wind speed	Climet - Model 011, 3 cup anemometer. Threshold 0.6 mph, distance constant <5 ft, operating range 0 to 110 mph, accuracy ±1% or 0.15 mph, whichever is greater	Esterline - Angus Model L825
	Wind direction	Wind direction	Climet - Model 012-10 wind vane. Threshold 0.75 mph, distance constant <3.3 ft, damping ratio 0.4	Esterline - Angus Model L825
	Temperature-differential $T_{300}-T_{33}$(1) $T_{150}-T_{33}$(2)			
	Dew point	Dew point	EG&G Model SM 110 accuracy +0.5°F	Westronics M11E
	Temperature-ambient	Temperature	Climet - Model 016-1 Motor-aspirated temperature shield with Climet 015-3 thermistor accuracy ±0.15°C	Leeds & Northrup Speedomax Multi-Point
6	Barometric pressure	Barometric pressure	Climet - Model 014-90 pressure transducer. Range 28-32 in. Hg	Esterline - Angus Model A
3	Rainfall	Rainfall	~~Model 0501 weighing rain gage~~ MRI MODEL 302 TIPPING BUCKET ACCURACY 0.01 INCHES	Esterline - Angus Model A

(1) Temperature taken as part of temperature differential measurement $T_{300} - T_{33}$.
(2) Temperature taken as part of temperature differential measurement $T_{150} - T_{33}$.
(3) Paired Climet 015-3 thermistor. Accuracy ±0.1°C.

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TABLE 2.3-29a

DATA ACQUISITION SYSTEM HARDWARE

MANUFACTURER	MODEL	QUALITY	DESCRIPTION
Hewlett Packard	9826A	2	Computer
Hewlett Packard	98256A	2	256K-Byte Memory Expansion
Hewlett Packard	98626A	4	Serial Ports
Hewlett Packard	3497	2	Data Acquisition/ Control Unit
Hewlett Packard	44421A	2	20-Channel Analog Multiplexor
Hewlett Packard	44425A	2	16-Bit Status Input
Dames & Moore	--	2	Transient Protection Modules (analog, status, voltage reference)
Hewlett Packard	9915	1	Computer
DEC	11/23	1	Computer
DEC	RX02	1	Disk Drive
DEC	VT 103BA	1	CRT
DEC		1	Serial Ports
Bell	212A	5	Modem (1)
Bell	202T	6	Modem (1)

(1) Or equivalent modem

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HCGS FSAR

TABLE 2.3-29b

SYSTEM MEASUREMENT ERROR

COMPONENT ERROR	WIND SPEED (10MPH)	WIND SPEED (30 MPH)	WIND SPEED (100 MPH)	WIND DIRECTION (DEGREES)	DELTA TEMPERATURE (300-33) (DEGREES CELSIUS)	DELTA TEMPERATURE (150-33) (DEGREES CELSIUS)	TEMPERATURE (DEGREES CELSIUS)	DEWPOINT (DEGREES CELSIUS)	PRECIPITATION (INCHES)
Sensor	± 0.15	± .30	± 1.00	± 3	± 0.10	± 0.10	± 0.10	± 0.5	.01
Translator	± 0.21	± 0.21	± 0.21	0	-	-	-	0.022	-
DVM	± 0.0035	± 0.0065	± 0.017	± 0.092	± 0.0026	± 0.0026	± 0.013	-	-
Software	0	0	0	0.00	0	0	0	0	-
Other	-	-	-	-	-	-	-	-	-
Total Maximum Error	0.3635	0.5165	1.227	± 3.092	± 0.1026	± 0.1026	± 0.113	± 0.522	.01
Root Sum Square Error	0.21	0.37	1.02	3.00	± 0.103	± 0.103	± 0.101	± 0.50	.01
R.G. 1.23 Specification	0.5	0.5	-	5.0	± 0.15	± 0.15	± 0.5	± 1.5	.01

(1) Instrumentation type and specification provided on Table 2.3-29 and 2.3-29a.

(2) The instantaneous error for wind speed measurements, assuming the individual component errors are additive and independent (root sum square error), is within the R.G. 1.23 specifications for all wind speeds less than 45 mph. The error of time averaged wind speeds will be less than the instantaneous root sum square error (this statement is applicable for all other parameters in this discussion). Therefore, for wind speeds considered to be most critical for dispersion calculations, the estimated error is well within the R.G. 1.23 specification.

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Table 2.3 - 29c

ARTIFICIAL ISLAND DIGITAL DATA ACQUISITION SYSTEM ACCURACIES

The following system accuracies are based upon VENDOR accuracy specifications and the following conditions:

- 1 year calibration interval
- 5-1/2 digits displayed on DVM
- Auto Zero ON

VOLTMETER ACCURACY

RANGE (V)	ERROR PERCENT OF READING	PLUS RESOLUTION ERROR (MV)
.119999	.015	.003
1.19999	.015	.02
11.9999	.015	.1

PARAMETER ERROR

PARAMETER	DVM RANGE	DAS INPUT VOLTAGE	DAS INPUT ENGINEERING UNITS	ERROR CALCULATION POINT	MAXIMUM[a] DAS ERROR
Temperature	1.19999V	0-1.0V	-30-+45°C	45°C	0.013°C
Delta-Temperature	1.19999V	0-1.0V	-5-+10°C	10°C	0.0026°C
Dew Point	11.9999V	0-5.0	-40-+100°F	100°F	0.022°F
Wind Speed	1.19999V	0-1.0V	0-100 mph	50 mph	0.0095 mph
Wind Speed	1.19999V	0-1.0V	0-100 mph	10 mph	0.0035 mph
Wind Speed	1.19999V	0-1.0V	0-100 mph	20 mph	0.0065 mph
Wind Direction	1.19999V	0-1.0V	0-540°	540°	0.092°
Precipitation	1.19999V	0-1.0V	0-1"	-	0.00"[b]
Pressure	1.19999V	0-1.0V	28-32"Hg	32Hg	0.00068"Hg
Solar Radiation	1.19999V	0-1.0V	0-2Ly/min	2Ly/min	0.00034Ly/min

[a]The data acquisition system error is due entirely to HP-3497A instrument error. Software calculations are computed to 12 significant digits. Therefore, software error is negligible.

[b]Precipitation is calculated using a step-function conversion technique with sufficient noise margin that an error of 0.00" is achievable over an entire calibration period interval.

DSER OPEN ITEM 3

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 Phenomena," 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)

can be made concerning compliances at this time.] -28g [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.

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.

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.

DSER OPEN ITEM 28a-c

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 comples	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

G6/3

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DSER Open Item No. 28G (Section 3.4.1)

FLOOD PROTECTION

The applicant has not provided the information requested concerning RG 1.102, position C.2, and therefore no conclusions can be made concerning compliances at this time.

RESPONSE

For the information requested above see response to Question 410.4.

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.

12 inch diameter

f. Scram discharge volume - The scram discharge volume (SDV) consists of two sets of header piping, each of which connects to one-half of the HCUs and drains into a 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.

12 inch diameter

The header piping slopes to a low point with a minimum pitch of 1/8" per foot as shown on Figure 4.6-1C

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.

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 acceptance of the BWROG position. HCGS will implement required fixes, if any, arising from NRC review and approval of the BWROG submittals by the end of the next refueling outage after NRC approval.

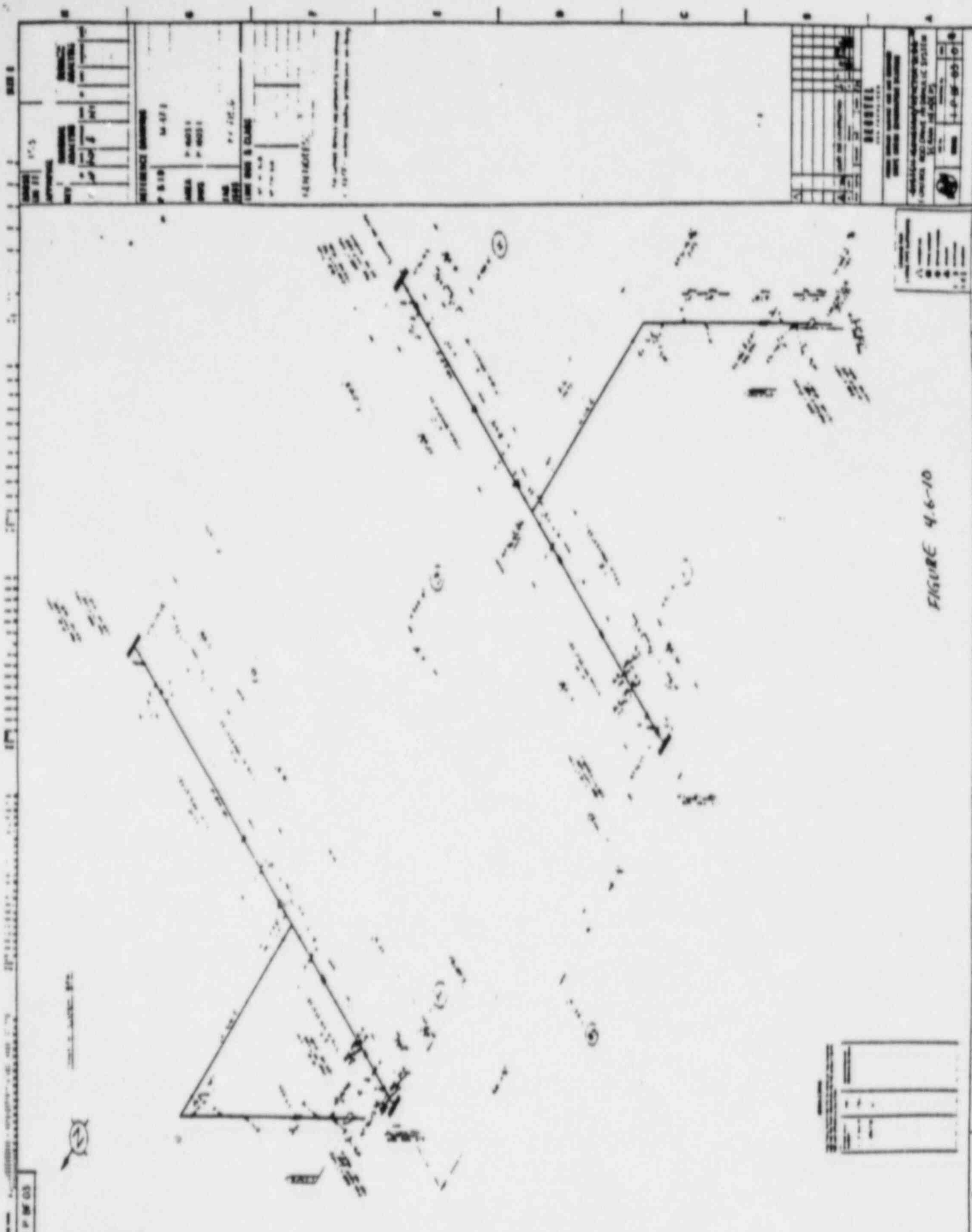

FIGURE 4.6-10

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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 capability 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.

DSER Open Item No. 112 (Cont'd)

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 three separate and diverse leak detection methods. The RCPB leak detection system consists of:

SEISMIC CATEGORY I QUALIFIED

and equipment

a. Drywell floor 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)

— INSERT D —

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^{-6} 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^{-6} to 10^{-4} 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

— INSERT D —

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.

Outflow from the sumps is not measured either, since outflow is not constant and is present only when the sump pumps are running. These pumps are started automatically when a predetermined high level in the sump is reached and pump at a constant rate until a low sump level signal automatically turns them off. A measurement of this outflow from the sumps would provide no useful information for leak detection purposes.

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.~~

— INSERT A —

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. ~~Co~~ These indicators ~~provide the operator~~ 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 ~~not necessary~~ 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 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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—INSERT B—

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

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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

— INSERT C —

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 ~~are~~ 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 flow rate 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.~~

— INSERT E —

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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-INSERT E-

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.

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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.

— INSERT F —

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

— INSERT F —

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.

DSER Open Item No. 121 (Section 6.2.1.3.3)

USE OF NUREG-0588

For the drywell, the limiting accident is a small-size break that does not result in reactor depressureization due to either loss of reactor coolant or automatic operation of the ECCS equipment. For this accident, the operators are alerted by a high drywell pressure signal and reactor scram, and it is assumed that they respond with an orderly shutdown of the reactor, which takes about 6 hours (i.e., the reactor coolant system is depressurized using the main condenser while limiting the reactor cooldown rate to 100°F/hr). The applicant has, therefore, assumed that there is a blowdown of reactor steam for the assumed 6-hour cooldown period. Because the worst combination of primary system pressure and drywell pressure produces a maximum superheat temperature of 340°F from the escaping steam for drywell design purposes and for the environmental qualification of safety-related equipment located in the drywell, the applicant has assumed a maximum drywell temperature of 340°F for 6 hours. We will require the applicant to comment on whether NUREG-0588 is being used for the temperature profile beyond 6 hours.

RESPONSE

In place of the temperature profile outlined in NUREG-0588, the temperature profile shown in Figure 3.11-4 of Section 3-11, is used for the environmental qualification of Class 1E equipment in the drywell as indicated in Section 6.2.1.1.2.6. This figure also shows the temperature used in the drywell beyond 6 hours for the environmental qualification of Class 1E equipment.

DSER Open Item No. 122 (DSER Section 6.2.1.3.3.)

TEMPERATURE PROFILE

The applicant has not provided the temperature profile to be used for environmental qualification of any safety-related equipment located in the suppression chamber. We will require that the applicant provide us with this information, and will report on this matter in a supplement to this SER.

RESPONSE

The safety-related components in the suppression chamber are the suppression pool-to-drywell vacuum breakers and Class 1E RTDs with associated cables. The RTDs and associated cables are qualified for drywell temperature. The vacuum breakers contain no 1E controls which would have to be qualified for post-LOCA suppression chamber temperature.

The vacuum breakers mechanical components are included in the program for qualification of mechanical equipment in harsh environments which is discussed in Section 3.11.2.6.

DSER Open Item No. 128 (Section 6.2.2)

AIR INGESTION

Ingestion potential has been extensively studied via full scale experiments, and BWR RHR suction/strainer geometries have been tested (see NUREG/CR-2772). Experimental results show that if the Froude (Fr) number is less than 0.8 at the intake, air ingestion is zero. We will require the applicant to comment on whether or not air ingestion poses a problem at HCGS.

RESPONSE

The HCGS RHR core spray, and HPCI suction strainer/piping geometries are such that the Froude number is less than 0.8 for all strainers. Therefore, air ingestion is not a concern for the HCGS design. For further discussion see Section 1.14.1.12 and revised Section 6.3.2.2.5.

1.14.1.10.6 Response (LRG II/2-RSB(d))

As indicated by FSAR Section 5.4.6.2.4(f), water hammer protection is provided for the RCIC system which is comparable to that provided for the ECCS injection systems.

1.14.1.11 <u>Adequate SRV Fluid Flow, LRG I/RSB-8</u>

1.14.1.11.1 Issue

The applicant must perform tests to show that flow through the safety relief valves is adequate to provide the necessary fluid relief required consistant with the analysis reported in Section 15.2.9 of the FSAR.

1.14.1.11.2 Response

See response to LRG Issue No. 5, Section 1.14.1.5.

1.14.1.12 <u>Provisions to Preclude Vortex Formation, LRG II/7-RSB</u>

1.14 1.12.1 Issue

To preclude vortex formation, air entrainment, and subsequent damage to ECCS pumps due to cavitation, it must be shown that adequate margin exists between the minimum suppression pool level and the depth of submergence of the ECCS pump suction strainers. This can be shown by analysis or by observations during pre-op testing that no vortex is formed.

1.14.1.12.2 Response

The ECCS pump suction strainers in the HCGS suppression chamber are provided with a minimum submergence of at least 10 feet, as measured from minimum suppression pool level. This amount of submergence ~~is expected~~ to provide sufficient margin to preclude formation of vortices, As indicated by FSAR Section 6.3.2.2.5, ~~the absence of air entrainment and vortex formation during ECCS pump operation will be verified during preoperational testing.~~

has been analyzed

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b. Instrumentation to indicate system performance during test operations

c. Motor-operated valves and check valves capable of manual operation for test purposes

d. Shutdown cooling lines taking suction from the recirculation system to permit testing of the RHR discharge into the RPV after normal plant shutdown

e. Drains to leak test the major system valves.

All active LPCI components are capable of individual functional testing during normal plant operation. Except as indicated below, the LPCI control system design provides automatic alignment from test to operating mode if system initiation is required. The exceptions are as follows:

a. Closure of any of the motor-operated pump suction valves in the suction lines from the suppression chamber requires operator action to reopen them. Indication of the status of these valves is provided in the main control room.

b. Parts of the system that are bypassed or deliberately rendered inoperative are indicated automatically or manually in the main control room.

6.3.2.2.5 ECCS NPSH Margin and Vortex Formation

NPSH calculations for ECCS pumps, such as the calculation in the previous section, have shown adequate margin to ensure capability of proper pump operation under accident conditions. This capability is verified during preoperational testing. ~~The absence of air entrainment and vortex formation during ECCS pump operation is also verified during preoperational testing.~~

insert →

6.3.2.2.6 ECCS Discharge Line Fill Network

A requirement of the ECCS is that cooling water flow to the RPV be initiated rapidly when the system is called upon to perform

Insert

The geometries of the RHR core spray and HPCI suction strainer and piping in the torus have been evaluated and the resulting Froude numbers are less than 0.8 for all strainers. Tests heve shown that no air core vortices or air withdrawal are observed for BWR Mark I geometries where the Froude number is less than 0.8. Therefore the HCGS design avoids the formation of air core vortices and possible air ingestion.

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<u>DSER Open Item No. 140</u> (DSER Section 9.1.2)

SPENT FUEL STORAGE

Since the applicant's application for an operating license was docketed in 1983, which is after the November 17, 1977 date specified in the SRP, the applicant must provide the results of an analysis which shows that a failure of the liner plate as a result of an SSE will not cause any of the following: (1) significant releases of radioactivity due to mechanical damage to the fuel; (2) significant loss-of-water from the pool which could uncover the fuel and lead to release of radioactivity due to heat up; (3) loss of the ability to cool the fuel due to flow blockage caused by a portion of one or more complete section of the liner plate falling on the top of the fuel racks; (4) damage to safety-related equipment as a result of the pool leakage; and (5) uncontrolled release of significant quantities on radioactive fluids to the environs; in accordance to the Standard Review Plan. These buildings are also designed against flooding and tornado missiles (refer to Section 3.4.1 and 3.5.2 of this SER). We cannot conclude that 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," Position C.3, 1.29, "Seismic Design Classification," Positions C.1 and C.2, have been met.

The applicant has not provided the design details of the spent fuel storage racks, the results of an analysis of impacts onto the racks, the bundle to bundle spacing, the design maximum enrichment (weight percent of U235), a description of calculational methods used for criticality analysis (along with the results), a tabulation of the nominal value of K_{eff} of the racks along with the various uncertainties and biases considered in the analysis, and a tabulation of the reactivity effect of each of the abnormal accident situations considered for our review. Since credit is taken for gadolinia in the fuel, the applicant must provide a commitment that every fuel bundle will have a specified minimum amount of gadolinia distributed over a specified number of specific fuel pins, for the entire length of the fuel. As an alternative, the applicant can provide the results of the criticality analysis without taking credit for the gadolinia.

Thus, we cannot conclude that the requirements of General Design Criteria 61, "Fuel Storage and Handling and Radioactivity Control," and 62, "Prevention of Criticality in Fuel Storage and Handling," and the guidelines of Regulatory Guide 1.13, Positions C.1 and C.4, concerning fuel storage facility design are satisfied.

DSER Open Item No. 140 (Cont'd)

We cannot conclude that the spent fuel storage facility is in conformance with the requirements of General Design Criteria 2, 61, and 62 as they relate to protection of the spent fuel against natural phenomena, radiation protection, and prevention of criticality and the guidelines of Regulatory Guides 1.13, Positions C.1, C.3, and C.4 and 1.29, Positions C.1 and C.2, relating to the facility's design basis and seismic classification. The spent fuel storage facility does not meet the acceptance criteria of SRP Section 9.1.2. 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 the evaluation of the compatibility and chemical stability of materials wetted by spent fuel pool water. To complete the review, the following information is requested:

(1) Identify and list all materials in the spent fuel storage pool including the neutron poison material, rack leveling feet, and rack frame.

(2) Provide test or operating data showing that the neutron poison material will not degrade during the lifetime of the spent fuel storage pool.

(3) Provide a description of any materials monitoring program for the pool. In particular, provide information on the frequency of inspection and type of samples used in the monitoring program.

(4) Provide details of the spent fuel racks to show that no buildup of gases will occur in the cavities containing the poison materials.

RESPONSE

The spent fuel pool liner plate was not designed to seismic Category I requirements because SRP 9.1.2, Revision 2 (March 1979), which first invoked the seismic Category I requirement, was not issued until after the design and procurement of the liner plate was complete and fabrication had begun (November 1978). However, the liner plate was designed to act as a form for the concrete in the spent fuel pool walls. To perform this function a system of channels, wide flanges and angle stiffeners was welded to the back surfaces of the liner and connected to the outside formwork with form ties. Thus, during the concrete placing operation the welds between the stiffeners and the liner were subject to the lateral pressure effects of the wet concrete. This may be considered a 'test' load in that after the concrete sets, the anchoring capability

RESPONSE (Cont'd)

of the stiffener system in holding the liner plate against seismic loads is at least equal to the form pressure load. The estimated test load during construction (approximately 300 lb/ft^2) was lower than the design value of 690 lb/ft^2. This construction load induced a correspondingly lower stress in the stiffener-to-liner welds.

An analysis, performed to evaluate the effect of SSE loads on the liner, shows that the resultant stresses would be insignificant (approximately 1% of the stresses due to concrete placement) when added to the residual concrete load. SSE induced loads imposed on the floor liner by the spent fuel racks would also be insignificant, and will not cause a liner failure.

Based on the considerable design margin for form pressure load and the acceptable performance of the wall liner plate when subjected to this 'test' load, it is concluded that the liner plate is capable of withstanding SSE loads without any loss of function.

Thus, the design of the liner plate satisfies General Design Criteria 2, 61, and 62, Regulatory Guide 1.29, Positions C.1 and C.2, and Regulatory Guide 1.13, Positions C.1 and C.4. Refer to Section 9.1.2.5 for additional justification of the non-seismic Category I liner design. For additional information on the design and analysis of the liner plate, refer to Appendix 3F.

For a discussion of the liner leakage collection system, which permits expedient liner leak detection and measurement, and prevents uncontrolled loss of contaminated pool water, refer to Section 9.1.2.2.2.1.

The spent fuel storage facility design meets the intent of Regulatory Guide 1.13 Position C.3, as described in Section 9.1.4.6 and 9.1.5.6.

The spent fuel storage rack design details have been provided in the response to Questions 281.2, 281.13, 410.39 and 410.42. The information requested in Questions 220.15 and 410.38 will be provided by September, 1984. This information will support the criticality review and demonstrate that the design satisfies General Design Criteria 61 and 62, and Regulatory Guide 1.13 positions C.1 and C.4.

The materials used in the spent fuel storage racks were included in the response to Question 281.13 (Amendment 5).

RESPONSE (Cont'd)

Similar rack designs, with vented Boral poison in stainless steel racks, have been licensed and have proven successful. HCGS's maximum anticipated radiation exposure for the Boral is 5.12×10^{11} rads. Similar Boral specimens have been subjected to accumulated radiation doses up to 7×10^{11} rads at the University of Michigan's Ford Ractor. These specimens were found to be structurally sound and neutron attenuation capabilities were not degraded by irradiation.

In order to continually assure the adequacy of the poison material, test coupons are provided for a Boral surveillance program. Forty-five coupons are installed in high radiation areas of the spent fuel pool. However, because stainless steel spent fuel racks with Boral poison material are already in use in other BWR fuel pools, a Boral surveillance program is not planned at HCGS.

If information from these lead plants indicates any problem with the Boral, a surveillance program can then be initiated.

The spent fuel rack poison cavities are vented to prevent any buildup of gases. Response to Question 281.13 provides further information on venting.

DSER Open Item No. 144 (DSER Section 9.2.1)

STATION SERVICE WATER SYSTEM

The SSWS consists of two redundant piping loops from the Delaware River to the plant. Each loop contains two 50% capacity pumps that are powered from a Class 1E power supply.

The system is housed in seismic Category I and tornato-protected structures (see Section 3.5.2 of this SER). [The applicant has not provided documentation to verify that the SSWS is protected from the flood water (including wave effects) of the design basis flood.] Station service water (SSW) piping is buried a minimum of 4 ft below grade, which provides adequate protection from missiles. The system is designed to seismic Category I, Quality Group C requirements. [Thus, we cannot conclude that the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," are satisfied.] However, the staff can conclude that the guidelines of RG 1.29, "Seismic Design Classification," Positions C.1 and C.2, are satisfied.

The design of the SSWS ensures that system function is not lost assuming a single active component failure coincident with a loss of offsite power. [However, the applicant has not demonstrated the design of the SSWS can provide sufficient cooling for a safe shutdown after a non-mechanistic pipe failure (event) with the loss of one SSWS pump (single active failure). Therefore, we cannot conclude that the requirements of General Design Criterion 44, "Cooling Water," are satisfied.]

The SSW pumps are normally operating. The availability of the standby pumps is ensured by periodic functional tests and inspections. The system design also incorporates provisions for accessibility to permit inservice inspection as required. [However, the applicant has not specified the frequency of the functional testing or inspection. Thus, we cannot conclude that the requirements of General Design Criteria 45, "Inspection of Cooling Water System" and 46, "Testing of Cooling Water System," are satisfied.]

[Based on the above, we cannot conclude that the station service water system meets the requirements of General Design Criteria 2, 44, 45, and 46, with respect to protection from natural phenomena, capability for transferring the required heat loads, inservice inspection and functional testing.] However, the staff concludes that the system meets the guidelines of RG 1.29, Positions C.1 and C.2, with respect to the system's seismic classification. [We will report resolution of this item in a supplement to this SER. The station service water system does not meet the acceptance criteria of SRP Section 9.2.1.]

RESPONSE

For information on the protection of the SSWS from flood water see the response to DSER Open Item No. 5.

Our response to Question 410.66 completely describes our design with regards to pipe break and loss of a service water pump. Briefly stated our design does not (and is not required according to BTP ASB 3-1, Section B.3.b.(3), for redundant trains of a dual-purpose moderate-energy essential system) consider non-mechanistic pipe breaks along with an additional single active failure of a pump.

9.2.1.6 Tests and Inspections

The system is hydrostatically tested prior to the station operation. All active components, e.g., pumps, valves, and controls, are functionally tested prior to startup and periodically thereafter. LEVEL AND FREQUENCY OF INSERVICE TESTING IS INCLUDED IN CHAPTER 16, TECHNICAL SPECIFICATIONS. Inservice Inspection and functional testing of the safety-related portions of the system and components will be in accordance with the examination and testing criteria of Articles IWA, IWD, IWP and IWV of Section XI, ASME Code, 1977 Edition and addenda through Summer, 1978.

The specific examination and tests of the system and components will be listed in the Station Inservice Inspection (ISI) and Inservice pump and valve test (ISI) program Administrative Procedures.

9.2.1.7 Instrumentation

Local instrumentation is provided at the equipment location for maintenance, testing, and performance evaluation.

Water levels at each station service water pump bay, and upstream of the intake structure, are monitored in the main control room. The station service water pump discharge header is equipped with pressure transmitters that provide input to the plant computer. Two dual element temperature sensors are located at opposite ends of the intake structure inlet. The river temperature displayed in the main control room is an average of these sensors.

9.2.2 SAFETY AND TURBINE AUXILIARIES COOLING SYSTEM

The safety and turbine auxiliaries cooling system (STACS) is a closed loop cooling water system consisting of two subsystems: a safety auxiliaries cooling system (SACS) and a turbine auxiliaries cooling system (TACS).

The SACS, which has a safety-related function, is designed to provide cooling water to the engineered safety features (ESF) equipment, including the residual heat removal (RHR) heat exchanger, during normal operation, normal plant shutdown, loss of offsite power (LOP), and a loss-of-coolant accident (LOCA).

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DSER Open Item No. 149 (DSER Section 9.3.3)

EQUIPMENT AND FLOOR DRAINAGE SYSTEM

The applicant has not provided an acceptable response to our concern of flooding due to a rupture of nonseismic Category I piping, vessels, or tanks, or due to the failure of a backflow prevention device in the drainage system. The ECCS compartments have seismic Category I water level instrumentation to alarm in the control room on high water level in the event of drain blockage flooding. The applicant has not provided the basis for not considering flooding after a safe shutdown earthquake which results in the worse case failure of the nonseismic Category I piping and only take credit for seismic Category I structures, systems, and components. Therefore, we cannot conclude that the system design meets the requirements of general Design Criteria 2, "Design Bases for Protection Against Natural Phenomena," and 4, "Environmental and Missile Basis," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," Positions C.1 and C.2, with respect to the failure of the drainage system resulting in equipment failure or in unacceptable release of radiation due to natural phenomena, missiles, or pipe breaks.

Based on our review, we cannot conclude that adequate protection against flooding of safety-related equipment and areas, and protection against the inadvertent release of potentially radioactive liquids to the environment through plant drainage paths is provided. We cannot, therefore, conclude that the system meets the requirements of General Design Criteria 2, 4, and 60, with respect to the need for protection against natural phenomena, pipe breaks, environmental effects (flooding), and release of radioactive material to the environment, and the guidelines of Regulatory Guide 1.29, Positions C.1 and C.2, with respect to seismic classification. The equipment and floor drain system does not meet the acceptance criteria of SRP Section 9.3.3. We will report resolution of this item in a supplement to this SER.

RESPONSE

Section 9.3.3.5 has been revised in response to Questions 410.91 and 410.93 to address the seismic qualification of the check valves and the plant safe shutdown capability following a SSE which results in a failure of the nonseismic Category I components and drain lines.

QUESTION 410.91 (SECTION (9.3.3)

Demonstrate that a failure of the nonseismic Category I, nonsafety-related portion of the equipment and floor drain system (EFDS) will not compromise the capability for safe shutdown because of failure of more than one redundant safety-related train due to flooding for the following reasons:

a. Failure of the EFDS to remove the flood water from an enclosure containing safety-related equipment. Consider flooding caused by a high energy pipe break, moderate energy pipe crack, and rupture of nonseismic Category I piping vessel or tanks;

b. Backflow in the EFDS due to check valve or other failure causing flooding of one safety-related enclosure from equipment or piping failure outside of this enclosure.

RESPONSE

The EFDS will not fail to remove flood water from enclosures containing safety-related equipment such that the capability to achieve safe shutdown would be compromised. Complete blockage or failure to pass flow of the EFDS is not considered credible and is not part of the design basis of HCGS. Blockage of a single EFDS line would preclude removal of flood water from the compartment served by that line. The HCGS design provides dedicated drain lines from safety related equipment compartments in the lower elevation of the plant to the sump to preclude cross flooding from one safety-related compartment to another. Also, to prevent flooding from one safety-related compartment to another, the walls, floors and penetrations are designed to be watertight. Section 9.3.3.5 has been revised to provide additional information on the floor drainage system.

As discussed above, significant flooding due to the failure of piping, equipment and instrumentation in the reactor building is not expected. However, in the event that significant quantities of water are conveyed to the sumps at elevation 54 feet, backflow into the ECCS compartments is prevented by the inclusion of a check valve where the dedicated drain line from each ECCS compartment terminates in the sump. Each ECCS compartment is provided with separate drain lines from the compartment to the sump. Thus, failure of any check valve will not result in flooding of more than one ECCS compartment. ~~An assessment is being performed to verify the ability of the check valves to maintain a functional pressure boundary against backflow following a SSE (See response to Question 410.93). This analysis will be completed by June 1984.~~ Section 9.3.3.5 has been revised to address seismic qualification of the check valves and the associated piping.

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system (RACS) water to keep the wastes at their normal operating temperature of 140°F.

9.3.3.4 System Operation

The equipment and floor drainage systems' wastes are selectively collected and drain directly to the area collection point by gravit. After collection in area sumps, the liquid radwaste is pumped to the radwaste collection tanks for processing by the appropriate treatment subsystems. The sump pumps start automatically when a preset high water level is reached. They stop at a preset low water level. Leaks inside the drywell drain to the drywell floor sump, except for the reactor recirculation pump seal leakoffs, which are routed to the drywell equipment sump. After a preset level is reached in the emergency sump in the turbine building and an alarm is annunciated, the sump contents are analyzed for radioactivity by recirculating through a sample loop before discharge.

The sanitary drainage system collects liquid wastes and entrained solids discharged by plumbing fixtures, with the exception of lavatory basins and showers in the personnel decontamination area and conveys them to the sewage treatment plant.

The storm drainage system collects water from precipitation on enclosure roofs, areaways, paved and unpaved surfaces, and irrigation runoffs outside the buildings, and conveys them to the Delaware River.

Low volume and oily water wastes from the emergency diesel generator and chemical regenerant waste from the makeup demineralizer, chemical storage tank dikes, equipment drains, transformer dikes, etc, are collected and pumped to the waste treatment plant in the yard area. These wastes are treated to a level that meets Environmental Protection Agency (EPA) and New Jersey Department of Environmental Protection (NJDEP) discharge limits before being discharged into the Delaware River.

9.3.3.5 Safety Evaluation

The plant drainage systems have no safety-related function. Failure of the system will not compromise any safety-related system or prevent a safe shutdown of the plant.

insert

ITem # 149

INSERT INTO FSAR SECTION 9.3.3.5.

PG 9.3-33

It has been verified that

~~F~~looding from a postulated failure of non-seismic, Category I systems and components (including large tanks and piping connected to these tanks) will not compromise the operation of any safety related sytem s or prevent safe shutdown of the plant. Most large volume non-seismic Category I tanks and systems are located in the turbine building and the radwaste area of the auxiliary building with no potential for flooding areas containing equipment required for safe shudown. Safety related cables in these areas are located above any potential ~~flow~~ flood levels.

In the reactor building and the control and diesel areas of the auxiliary building potential flooding from postulated failure of non-seismic Category I systems and components is contained within the compartment containing the equipment. The flooding will drain to the respective floor drain sump. Essential equipment is located in areas not subject to flooding by the failure of nonseismic Category I components or in compartments that are protected from flooding from sources extended to the compartment.

In the unlikely event that a seismic event also causes the exposed drain line from the postulated flood area to leak, the fluid may drip into an area containing essential equipment.

The essential equipment is either located such that it is not subjected to the dripping or is designed to withstand the effects of the dripping. The dripping fluid will drain from the compartment through the floor drains.

ITem # 149

F67(1)

QUESTION 410.93 (SECTION (9.3.3)

Verify that all check valves which protect safety-related equipment from flooding due to backflow through the drainage systems are seismic Category I.

RESPONSE

~~All the check valves which protect safety-related equipment from flooding due to backflow through the drainage systems will be seismically qualified. Seismic qualification will be completed by June 1984.~~

Section 9.3.3.5 has been revised to address the seismic qualification of the check valves and the associated piping.

ITem # 149

Each emergency core cooling system (ECCS) compartment is provided with a separate drain line to the reactor building DRW sump. Flooding of the ECCS pump compartments in the reactor enclosure by backflow through the floor drains and equipment drain funnels is prevented by the use of adjustable check valves (backwater) installed in these lines. A normally closed manual valve is provided for the floor drain line in the safety auxiliary cooling system (SACS) pump compartment to prevent backflow. The adjustable check valves are evaluated to assure they will perform their function following an SSE. The piping inside the sumps associated with these check valves has been analyzed to assure it will not fail in an SSE. Each ECCS compartment is equipped with watertight doors to prevent any spread of the flooding. In the ECCS compartment, Seismic Category I level instrumentation installed in the main control room for high water level alarms in the event of drain blockage or flooding.

The drywell drain sumps and the floor drain sumps in the reactor enclosure are also used as a means to detect plant leakage as discussed in Section 5.2.5.

9.3.3.6 Tests and Inspections

All drainage piping is tested prior to its embedment in concrete. Potentially radioactive drainage piping is pneumatically tested to 20±5 psig air for a minimum of 10 minutes, in accordance with ANSI B31.1 (1973). Nonradioactive oily, acid, and storm drainage piping is hydrostatically tested to the equivalent of 20 ±5 psig for 10 minutes. The sanitary drainage piping is tested according to the National Standard Plumbing Code at a hydrostatic pressure of 10 feet of water for 15 minutes. Plant drainage system operability is checked by normal use and by the instrumentation provided in the sumps and the main control room.

9.3.3.7 Instrumentation Application

a. Drywell equipment and floor drain sumps - A level measurement in each sump is fed to a local radiation processor that starts and stops the lead sump pump at a preset high and low levels, respectively. The processor also starts the second pump and alternates the lead pump after each pump cycle. The alarm on high level in each sump is annunciated in the main control room.

Item # 149

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 satistied. 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

DSER Open Item No. 151 (Cont'd)

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 2.2.3.13.

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.

DSER Open Item No. 151 (Cont'd)

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 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 (approximately 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 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 Tests and Inspections

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.

DSER Open Item 176c (Section 14.2)

INITIAL PLANT TEST PROGRAM

Provide a response to Q640.9.

RESPONSE

The complete response to Q640.9 was provided as part of Amendment 6 to the HCGS FSAR.

DSER Open Item 176d (Section 14.2)

INITIAL PLANT TEST PROGRAM

The response does not address the concerns of I&E Information Notice Number 83-17, March 31, 1983. The concern is that if a time delay prevents fuel from being supplied to the diesel generator following a shutdown signal, the air supply may be exhausted before the fuel supply is reinstated. The response to this item should be modified to address these concerns.

RESPONSE

The response to Q640.10 has been revised in Amendment 6 to the HCGS FSAR to provide the information requested above.

DSER Open Item 176e (Section 14.2)

INITIAL PLANT TEST PROGRAM

Provide response to Q640.11 Item 2.

RESPONSE

The information requested above was provided as part of Amendment 3 to the HCGS FSAR.

DSER Open Item 176i (Section 14.2)

INITIAL PLANT TEST PROGRAM

Provide response to Q640.21 items 4, 5, and 6.

RESPONSE

The information requested above was provided as part of Amendment 3 of the HCGS FSAR.

DSER Open Item No. 184 (Section 7.2.2.1)

FAILURE IN REACTOR VESSEL LEVEL SENSING LINES

The applicant is required to submit the results of the analysis concerning failures in reactor vessel level sensing lines to the NRC for review and provide a description of the proposed modifications or justify why modifictions are not necessary.

RESPONSE

For the information requested above, see the response to Question 421.23.

DSER Open Item No. 206 (Section 7.6.2.1)

HIGH PRESSURE/LOW PRESSURE INTERLOCKS

The applicant was asked to discuss the design details utilized at HCGS for overpressurization protection of the low pressure ECCS. In response, the applicant provided acceptable design details for the ECCS high pressure/low pressure interlocks. However, the staff remained concerned regarding the setpoints utilized for these interlocks.

The applicant is required to provide the design basis for the selection of the setpoints utilized for ECCS high pressure/low pressure interlocks.

RESPONSE

Design details for the ECCS high pressure/low pressure interlocks are presented in the response to Questions 440.21 and 440.26 and are summarized in the response to DSER Open Item No. 135 (DSER Section 6.3.3). As these responses describe, overpressurization protection for the RHR low pressure piping is provided by the LPCI injection check valve rather than by differential pressure interlocks on the LPCI injection valves. Hence, there are no LPCI pressure interlock setpoints.

The core spray system injection (isolation) valves are interlocked directly with reactor pressure. A pressure indicating switch, N690 (see Figure 6.3-7) with a nominal setpoint of 461 psig and an allowable value of 441 psig, provides an opening permissive signal when the reactor pressure falls below the maximum design pressure (approximately 460 psig) for the core spray discharge piping.

HOPE CREEK

DSER OPEN ITEMS 211a, b, d; 212, 213, 214, 215, 216a

The main concern is that the applicant's alternative approaches to RGs 1.37, "Quality Assurance Requirements for Cleaning Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and 1.44, "Control of the Use of Sensitized Stainless Steel," do not provide an acceptble level of protection from intergranular stress corrosion cracking . . . For Regulatory Guide 1.44, the applicant has set high chloride content limits that exceed the recommendations of the guide. The applicant's high chloride limit of 200 ppm and the chloride limits for other materials that come in contact with austenitic stainless steels do not provide protection from concentrations of chlorides that can occur by evaporation. The same situation applies to the 100-ppm limit for chloride content of the final flushing water.

Cleaning and cleanliness control are not in accordance with the recommendations of RG 1.37, "Quality Assurance Requirements for Cleaning Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants." The chloride content limits for flushing fluids are too high and are not acceptable to the staff.

RESPONSE

In Regulatory Guide 1.44 reference is made to Regulatory Guide 1.37 for the quality of water for cleaning and flushing of fluid systems. Regulatory Guide 1.37 further references ANSI N45.2.1.-1973 as an acceptable basis for complying with the pertinent quality assurance requirements of Appendix B to 10CFR Part 50.

For the NSSS and non-NSSS scope of supply the requirements specified in the applicable GE and Bechtel specifications for cleanness of piping and equipment are in strict compliance with Regulatory Guide 1.37 and ANSI N45.2.1-1973 regarding the water quality requirements of freshwater and demineralized water for rinsing and flushing purposes.

For non-metallic materials that come in contact with austenitic stainless steel, such as die lubricants, marking materials, masking tape, cleaning solutions, etc., the GE and Bechtel specifications require that the chloride concentrations be controlled in accordance with the various relevant Regulatory Guides and ANSI standards. Further these materials are removed and the surfaces cleaned and rinsed immediately following the operation in which they are used. Since the quality of the rinse and flush water is being

maintained there is adequate protection from concentrations of chlorides that could occur by evaporation.

FSAR Section 1.8.1.44 has been reviewed to the applicable GE and Bechtel specifications. This review resulted in the revision of Position C1 to provide clarification of several statements and the deletion of references to the use of trichlorotrifluoroethane (TCTFE), which is prohibited, such that this section more accurately describes the actual practice.

are controlled so that halogen and sulfer levels agree with the various Regulatory Guides or ANSI standards covering these materials. In addition, these materials are removed immediately following the operation in which they are used and prior to any elevated-temperature treatment.

1.8.1.44 Conformance to Regulatory Guide 1.44, Revision 0, May 1973: Control of the Use of Sensitized Stainless Steel

HCGS complies with Regulatory Guide 1.44, except as noted below.

Architect-engineer-procured items and architect-engineer field work comply with Regulatory Guide 1.44, subject to exceptions or clarifications stated below that are applied to ASME B&PV Code, Section III equipment and piping in safety-related systems. They are not generally applied to HVAC systems or to instruments.

in accordance with Regulatory Guide 1.37 and ANSI N45.2.1-1973.

Position C.1 of Regulatory Guide 1.44 is complied with since contamination of austenitic stainless steel (Type 300 series) by compounds that could cause stress corrosion cracking is avoided during all stages of fabrication and installation. ~~Except for trichlorotrifluoroethane (TCTFE) meeting the requirements of Military Specification MIL-C-81302B, cleaning is limited to solutions that contain not more than 200 ppm of chlorides. Rinsing or flushing is done with water that contains not more than 200 ppm of chlorides. Special rinsing techniques are used to ensure complete removal of TCTFE where crevices or undrainable areas occur.~~

Nonmetallic materials ~~Foreign substances~~ in contact with austenitic stainless steel, ~~i.e., die lubricants, penetrant materials, marking materials, masking tape, etc, are controlled so that they contain no more than 200 ppm of chlorides, or they are removed immediately following the operation in which they were used.~~ Penetrant materials may conform to the higher contaminant levels specified in Article 6, Section V, of the ASME B&PV Code, provided that the materials are thoroughly removed immediately after the examination has been completed. and the surface cleaned Crevices and ~~undrainable areas~~ ~~are protected prior to the use of materials containing more than 200 ppm of chlorides. All substances in contact with austenitic stainless steel are removed prior to any elevated temperature treatment.~~ small openings are protected from contamination.

Completed components are packaged ~~in~~ such ~~a manner~~ that they are protected from the weather, dirt, wind, water spray, and any other extraneous environmental conditions that may be encountered during shipment and subsequent site storage.

In the field, austenitic stainless steel components are stored clean and dry. Components either are stored indoors, or, if outdoors, are stored off the ground and covered with tarps.

Contamination of austenitic stainless steels in the field during installation is avoided as described above. The system hydrostatic test and the preoperational testing and final flushing of the completed system is performed with equivalent to reactor coolant grade water ~~that contains not more than 100 ppm of chlorides.~~ Nonmetallic insulation composed of leachable chloride and fluoride materials that come into contact with austenitic stainless steel are held to the lowest practicable level by the inclusion of the requirements of Regulatory Guide 1.36 in the insulation purchase specifications.

Position C.2 of Regulatory Guide 1.44 is complied with since all grades of austenitic stainless steels (Type 300 series) are required to be furnished in the solution heat-treated condition before fabrication or assembly into components or systems. The solution heat treatment varies according to the applicable ASME or ASTM material specification.

Position C.3 of Regulatory Guide 1.44 covers all austenitic stainless steels furnished in the solution heat-treated condition in accordance with the material specification. During fabrication and installation, austenitic stainless steels are not permitted to be exposed to temperatures in the range of 800 to 1500°F, except for welding and hot forming. Welding practices are controlled to avoid severe sensitization, and solution heat treatment in accordance with the material specification is also required following hot forming in the temperature range of 800 to 1500°F. Unless otherwise required by the material specification, the maximum length of time for cooling from the solution heat-treated temperature to below 800°F is specified in the equipment specification. Corrosion testing in accordance with ASTM A 262-70, Practice A or E, may be required if the maximum length of time for cooling below 800°F is exceeded, or the solution heat-treated condition is in doubt.

No austenitic stainless steel is subjected to service temperatures in the range of 800 to 1500°F, as discussed in Position C.4 of Regulatory Guide 1.44. The only exposure of austenitic stainless steels to this range of temperatures occurs on the containment hydrogen recombiner system (CHRS) and subsequent to solution heat-treating during welding. Welding practices are controlled as discussed below. In addition, the architect-engineer-supplied austenitic stainless steel piping and

DSER Open Item 211c (Section 4.5.1)

CONTROL ROD DRIVE STRUCTURAL MATERIALS

The allowed welding heat input limit of 100 kj/in for the fabrication of control rod drive components has been shown by General Electric to sensitize Type 304 austenitic stainless steel and accordingly is unacceptable.

RESPONSE

The welding specification controlling the fabrication of control rod drive (CRD) components at GE's Wilmington, NC manufacturing operations has always specified a heat input limit of 50 Kj/in. The HCGS CRD components were fabricated under this specification. Section 4.5.1.2.1 has been revised to remove the reference to the description of compliance to Regulatory Guide 1.44 in Section 4.5.2.4.4, which deals with reactor vessel internals.

a. The cylinder and spacer (cylinder, tube and flange assembly) and the retainer (collet assembly) are hard surfaced with Colmonoy 6.

b. The following components are nitrided to provide a wear resistant surface:

   1. Piston tube (piston tube assembly)

   2. Index tube (drive line assembly)

   3. Collet piston and guide cap (collet assembly).

Colmonoy hard surfacing is applied on the cylinder, spacer, and retainer by the flame spray process.

Nitriding is accomplished using a proprietary process called New Malcomizing. Components are exposed to a temperature of about 1080°F for approximately 20 hours during the nitriding cycle.

Colmonoy hard surfaced components have performed successfully for the past 20 years in drive mechanisms. Nitrided components have been used in CRDs since 1967. It is normal practice to remove some CRDs at each refueling outage. At this time, both the Colmonoy hard surfaced parts and the nitrided surfaces are accessible for visual examination. In addition, dye penetrant examinations have been performed on nitrided surfaces of the longest service drives. This inspection program is adequate to detect any incipient defects before they can become serious enough to cause operating problems.

Welding is performed in accordance with Section IX of the ASME B&PV Code. Heat input for stainless steel welds is restricted to a maximum of 50,000 Joules per inch and an interpass temperature of 350°F. These controls are employed to avoid severe sensitization and comply with the intent of Regulatory Guide 1.44. ~~For general compliance or alternate approach assessment for Regulatory Guide 1.44, see Section 4.5.2.4.4.~~

DSER OPEN ITEM 211C

4.5.2.4.2 Conformance with Regulatory Guide 1.34, Control of Electroslag Weld Properties

Electroslag welding is not employed for any reactor internals.

4.5.2.4.3 Conformance with Regulatory Guide 1.36, Nonmetallic Thermal Insulation for Austenitic Stainless Steel

For external applications, all nonmetallic insulation meets the requirements of Regulatory Guide 1.36.

4.5.2.4.4 Conformance with Regulatory Guide 1.44, Control of the Use of Sensitized Stainless Steel

All wrought austenitic stainless steel is purchased in the solution heat treated condition. Heating above 800°F is prohibited (except for welding) unless the stainless steel is subsequently solution annealed. For 304 stainless steel with carbon content in excess of 0.035% carbon, purchase specifications restrict the maximum weld heat input to 1~~1~~00,000 Joules per inch, and the weld interpass temperature to 350°F maximum. Welding is performed in accordance with Section IX of the ASME B&PV Code. These controls are employed to avoid severe sensitization, and comply with the intent of Regulatory Guide 1.44.

4.5.2.4.5 Conformance with Regulatory Guide 1.71, Welder Qualification for Areas of Limited Accessibility

There are few restrictive welds involved in the fabrication of items described in this section. Mock-up welding is performed on the welds with most difficult access. Mock-ups are examined with radiography or by sectioning.

4.5.2.4.6 Conformance with Regulatory Guide 1.37, Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants

Exposure to contaminants is avoided by carefully controlling all cleaning and processing materials that contact stainless steel

DSER ITEM 211C

DSER Open Item No. 211e (Section 4.5.1)

CONTROL ROD DRIVE STRUCTURAL MATERIALS

The applicant should identify the materials specifications used in the control rod drive components made of ARMCO 17-4 PH, and Inconel X-750.

RESPONSE

The fingers of the collet assemblies and the coupling spuds of the drive line assembli s of the HCGS control rod drives (CRDs) were fabricated of Inconel X-750, which was specified by a General Electric specification similar to ASTM A637, G688, Type 2. The collet springs of the CRDs were fabricated of Inconel X-750, which was specified by a General Electric specification similar to AMS 5699. The piston heads of the drive line assemblies were fabricated of 17-4 PH, which was specified by a General Electric equivalent to ASTM A564, Type G630 with a 1100°F age hardening.

DSER Open Item No. 216b (Section 5.3.1)

REACTOR VESSEL MATERIALS

The reactor vessel studs and fasteners satisfy some of the recommendations of RG 1.65, "Materials and Inspections for Reactor Vessel Closure Studs." The FSAR does not discuss the nondestructive examinations of the stud bolts and nuts.

RESPONSE

The main closure studs, nuts, and washers for the reactor vessel are ultrasonically examined in accordance with Paragraph N-322 of Section III of the ASME B&PV Code and additional GE requirements. Magnetic particle inspections of the surfaces of the main closure studs, nuts and washers, are conducted in accordance with Paragraph N-626 of Section III of the ASME B&PV Code.