PSEG Public Service Electric and Gas Company

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Robert L. Mittl General Manager
Nuclear Assurance and Regulation

July 18, 1984

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

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

Gentlemen:

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

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

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

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

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

Very truly yours,

R L Mittl

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Attachments

C D. H. Wagner
USNRC Licensing Project Manager

W. H. Bateman
USNRC Senior Resident Inspector

FM05 1/2

DATE: 7/18/84

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	Open	
2b	2.3.3	Accuracies of meteorological measurements	Open	
2c	2.3.3	Accuracies of meteorological measurements	Open	
2d	2.3.3	Accuracies of meteorological measurements	Open	
3a	2.3.3	Upgrading of onsite meteorological measurements program (III.A.2)	Open	
3b	2.3.3	Upgrading of onsite meteorological measurements program (III.A.2)	Open	
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	

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	Int-ke structure sliding stability	Open	
27	2.5.5	Slope stability	Complete	6/1/84
28a	3.4.1	Flood protection	Open	
28b	3.4.1	Flood protection	Open	
28c	3.4.1	Flood protection	Open	
28d	3.4.1	Flood protection	Open	
28e	3.4.1	Flood protection	Open	
28f	3.4.1	Flood protection	Open	
28g	3.4.1	Flood protection	Open	
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

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 LOCA 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	Open	
110b	4.6	Functional design of reactivity control systems	Complete	6/1/84
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	Open	
112b	5.2.5	Reactor coolant pressure boundary leakage detection	Open	
112c	5.2.5	Reactor coolant pressure boundary leakage detection	Open	
112d	5.2.5	Reactor coolant pressure boundary leakage detection	Open	
112e	5.2.5	Reactor coolant pressure boundary leakage detection	Open	
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	Open	
122	6.2.1.3.3	Temperature profile	Open	
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	Open	
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	Open	
140b	9.1.2	Spent fuel pool storage	Open	
140c	9.1.2	Spent fuel pool storage	Open	
140d	9.1.2	Spent fuel pool storage	Open	
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	6/29/84
141e	9.1.3	Spent fuel pool cooling and cleanup system	Open	6/29/84
141f	9.1.3	Spent fuel pool cooling and cleanup system	Open	6/29/84
141g	9.1.3	Spent fuel pool cooling and cleanup system	Complete	6/15/84
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 drainage system	Open	
149b	9.3.3	Equipment and floor drainage system	Open	
150	9.3.6	Primary containment instrument gas system	Open	
151a	9.4.1	Control structure ventilation system	Open	
151b	9.4.1	Control structure ventilation system	Open	
152	9.4.4	Radioactivity monitoring elements	Closed (5/30/84-Aux.Sys.Mtg.)	6/1/84
153	9.4.5	Engineered safety features ventilation system	Open	
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	

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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	Open	
176d	14.2	Initial plant test program	Open	
176e	14.2	Initial plant test program	Open	
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	Open	
177	15.1.1	Partial feedwater heating	Complete	7/18/84
178	15.6.5	LOCA resulting from spectrum of postulated piping breaks within RCP	NRC Action	
179	15.7.4	Radiological consequences of fuel handling accidents	NRC Action	
180	15.7.5	Spent fuel cask drop accidents	NRC Action	
181	15.9.5	TMI-2 Item II.K.3.3	Complete	6/29/84
182	15.9.10	TMI-2 Item II.K.3.18	Complete	6/1/84
183	18	Hope Creek DCRDR	Open	
184	7.2.2.1.e	Failures in reactor vessel level sensing lines	Open	
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	

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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	Open	
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	Open	
211b	4.5.1	Control rod drive structural materials	Open	
211c	4.5.1	Control rod drive structural materials	Open	
211d	4.5.1	Control rod drive structural materials	Open	
211e	4.5.1	Control rod drive structural materials	Open	
212	4.5.2	Reactor internals materials	Open	
213	5.2.3	Reactor coolant pressure boundary material	Open	
214	6.1.1	Engineered safety features materials	Open	
215	10.3.6	Main steam and feedwater system materials	Open	
216a	5.3.1	Reactor vessel materials	Open	

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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	Open	
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	8.3.3.3.3	Specified separation of raceways by analysis and test	Open	
247	8.3.3.5.1	Capability of penetrations to withstand 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	4.4.4	Stability analysis submitted before second-cycle operation	Open	

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ATTACHMENT 2 DATE: 7/18/84

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			

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ATTACHMENT 3

OPEN ITEM	DSER SECTION NUMBER	SUBJECT
29	3.5.1.1	Internally Generated Missiles (Outside Containment)
31	3.5.1.3	Turbine Missiles
34	3.6.2	Unrestrained Whipping Pipe Inside Containment
90	3.9.2.1	Vibration monitoring program during testing
105	4.2	Plant-specific mechanical fracturing analysis
106	4.2	Applicability of seismic and LOCA loading evaluation
113	5.3.4	GE Procedure Applicability
114	5.3.4	Compliance with NB 2360 of the Summer 1972 Addenda to the 1971 ASME Code
115	5.3.4	Drop Weight and Charpy V-Notch Tests for Closure Flange Materials
116	5.3.4	Charpy V-Notch Test Data for Base Materials as Used in Shell Course No. 3
127	6.2.1.6	Operability Testing of Vacuum Breakers
131	6.2.3	Administration of secondary containment openings

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ATTACHMENT 3 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT
136	6.3.5	Plant-specific LOCA (also Section 15.9.13)
166	12.3.4.2	Airborne radioactivity monitor positioning
167	12.3.4.2	Portable continuous air monitors
169	12.5.3	Guidance of Division 8 Regulatory Guides
177	15.1.1	Partial feedwater heating

ATTACHMENT 4

DSER Open Item No. 29 (Section 3.5.1.1)

INTERNALLY GENERATED MISSILES (OUTSIDE CONTAINMENT)

With respect to rotating equipment, the applicant has stated that the pumps and fans were manufactured to the same industry standards as Palo Verde and therefore the results of the Palo Verde's analysis for internally generated missiles is applicable to Hope Creek. In order to rely upon the analysis performed by Palo Verde, the applicant must verify that every rotating component (pumps, fans, motors, and turbines, except the main turbine-generator) is designed and constructed to exactly the same codes and standards (including addenda and editions), to be of the same manufacturer, size, and materials as the analyzed components at Palo Verde. Palo Verde relied mainly upon compartmentalization as the means to protect the redundant equipment. For each component where compartmentalization was relied upon at Palo Verde, the applicant must verify the identical components at Hope Creek provided with comparable compartmentalization.

Similarly, the applicant must verify the use of barriers, separation and orientation as was used by Palo Verde. For every component which is not identical with Palo Verde, the applicant must provide a discussion of the analysis which verifies that the casing would be capable of retaining the internally generated missile or that the missile would not strike safety-related components or generate a secondary missile. Unless the applicant either verifies comformance with the Palo Verde design (as outlined above) or provides the results of an analysis which shows that the casings will contain the internally generated missiles, the applicant must provide protection by any one or a combination of compartmentalization, barriers, separation, orientation, and equipment design. Safety-related systems must be verified to be physically separated from nonsafety-related systems and components of safety-related systems are physically separated from their redundant compartments.

Based on the above, we cannot conclude that the design is in conformance with the requirements of General Design Criterion 4 as it relates to protection against internally generated missiles until the applicant provides an acceptable discussion concerning rotating components as potential sources of internally generated missiles. We cannot determine that the design of the facility for providng protection from internally generated missiles meets the applicable acceptance criteria of SRP Section 3.5.1.1. We will report resolution of this item in a supplement to this SER.

RESPONSE

FSAR Section 3.5.1.1 has been revised to include the results of an analysis of the internally generated rotational missiles outside containment.

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DSER OPEN ITEM 29

CHAPTER 3

TABLES

CHAPTER 3

TABLES (cont)

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3.5 MISSILE PROTECTION

The Seismic Category I and safety-related structures, equipment, and systems are protected from postulated missiles through basic plant arrangement so that a missile does not cause the failure of systems that are required for safe shutdown or whose failure could result in a significant release of radioactivity. Where it is impossible to provide protection through plant layout, suitable physical barriers are provided to shield the critical system or component from credible missiles. Redundant safety-related Seismic Category I components are arranged so that a single missile cannot simultaneously damage a critical system component and its backup system.

A tabulation of safety-related structures, systems, and components, their locations, seismic category, quality group classification, and the applicable FSAR sections is given in Table 3.2-1. General arrangement drawings are included as Figures 1.2-2 and 1.2-41.

3.5.1 MISSILE SELECTION AND DESCRIPTION

3.5.1.1 Internally Generated Missiles (Outside Primary Containment)

The systems located outside the primary containment have been examined to identify and classify potential missiles. These systems and missiles are listed in TableS 3.5-1 and 3.5-13. Redundant systems are normally located in different areas of the plant or separated by missile-proof walls so that a single missile can not damage both systems.

~~Large pumps, such as~~ The residual heat removal (RHR) and core spray pumps, are located in separate missile-proof compartments and their impellers are enclosed in a concrete structure therefore they ~~and~~ are not considered a potential missile source or hazard to other systems.

Refer to Section 3.5.3 for barrier design procedure.

There are three general sources of postulated missiles:

a. Rotating component failure

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b. Pressurized component failure

c. Gravitationally generated missiles.

3.5.1.1.1 Rotating Component Failure Missiles

Catastrophic failure of rotating equipment having synchronous motors, e.g., pumps, fans, and compressors, that could lead to the generation of missiles is not considered probable ~~credible.~~ Massive and rapid failure of these components is improbable because of the conservative design, material characteristics, inspections, and quality control during fabrication and erection. Also, the rotational speed is limited to the design speed of the motor, thereby precluding component failures due to runaway speeds.

Similarly, it is concluded that the high pressure coolant injection (HPCI) and reactor core isolation cooling (RCIC) pumps and turbines cannot generate credible missiles. These pumps are not in continuous use, but are periodically tested and otherwise operate only in the unlikely event of a postulated accident. They are classified as moderate energy systems. Overspeed tripping devices ensure that the turbines do not reach runaway speed, where failure leading to the ejection of a missile could take place.

~~Other rotating equipment does not constitute a missile hazard because of its small size and/or the unlikelihood that its rotating components would penetrate its housing~~.

Insert 1 →

3.5.1.1.2 Pressurized Component Failure Missiles

The following are potential internal missiles from pressurized equipment:

a. Valve bonnets

b. Valve stems

c. Temperature detectors

d. Nuts and bolts

INSERT 1

A tabulation of missiles generated by postulated failures of rotating components, their sources and characteristics, and a safety evaluation are provided in Table 3.5-13.

The evaluation identified one instance where a postulated missile, which could penetrate through the flexible connection of a vane-axial fan, could have the potential to damage safe-shutdown equipment in the room. In order to prevent the postulated missile from damaging safety-shutdown equipment, a missile shield has been added to the design to withstand the impact of the postulated fan blade missile.

The formulas used to predict the penetration resulting from missile impact are provided in Reference 3.5-4. The penetration and perforation formulas assume that the missile strikes the target normal to the surface, and the axis of the missile is assumed parallel to the line of flight. The rotating components is assumed to fail at 120 percent overspeed. These assumptions result in a conservative estimate of local damage to the target.

TABLE 3.5-1 PRESSURIZED COMPONENT

INTERNALLY GENERATED PRESSURIZED COMPONENT MISSILES OUTSIDE CONTAINMENT

System	FSAR Section	Missile Description	Protection Evaluation Codes[a]
HPCI	6.3	Test connection	c
		Startup flange	c
		Pressure indicator (PI-R003)	c
CRD hydraulic	4.6.1	Drains	c
		Pressure indicators (PI-R008, 4013 A, B)	c
		Pressure indicators (PI-R021, PI-N005, PI-R016, PI-R012, PI-R007, PI-R010, PI-R006)	c
		Test indicators (TI-4014, TE-4014, TE-N018)	c
		Test connections	c
		Vent	d
		Blind flange	c
Main steam	5.1	Test connections	c
		Temperature elements (TE-N040)	c
		Pressure indicators (PP-3632 A, B, C, D)	c
Main steam sealing	5.1	Temperature elements (TE-N057 A, B, C, D, E)	c
		Pressure transmitter (PT-5838)	c
		Blind flange or Y-strainer	c
		Test connection	c
		Temperature element (TE-N060)	c
Feedwater	5.1	Test connection	c
RWCU	5.4.8	Blind flange	c
		Temperature sensors/elements (TE-N007, TE-N019, TE-N015, TE-N004, TS-169, TS-170, TS-2[illegible] A, B)	c
		Pressure transmitter (PT-N005)	c
		Pressure point (PP-3876 A, B; PP-3875 A, B; PP-3916 A, B; PP-3917 A, B)	c
RWCU	5.4.8	Pressure indicators (PI-3377 A, B; PI-R009; PI-R004; PI-R008; PDIS-3987 A, B; PDIS-3988 A, B)	c
		Pressure switches (PSL-N013, PSH-N014)	c
		Flow elements (FE-3986 A, B)	c

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TABLE 3.5-13

INTERNALLY GENERATED ROTATING COMPONENT
MISSILES OUTSIDE CONTAINMENT

MISSILE IDENTIFICATION	SOURCE OF MISSILE	LOCATION	MISSILE CHARACTERISTICS VELOCITY (FT/S)	DIA. (IN.)	WEIGHT (LBs)	CALCULATED MAX. STEEL PERF. DEPTH (IN.)	CASING THICKNESS	REMARKS
Fan Blade	Containment Pre-purge Cleanup Fan 10V-200 (Centrifugal Fan)	Reactor Bldg El. 162'	199.0	1.21	3.7	0.211	0.1406	Fan blade may penetrate fan casing. The surrounding concrete wall for the fan is 12" thick. The calculated depth of fan blade penetration into the concrete wall is 1.43". Therefore, missile has no effect on plant safe shutdown capability. Therefore protection is not needed.
Fan Blade	Diesel Generator Wing Area Exhaust Fan 1A,B-V414 (Centrifugal Fan)	Aux Bldg SDG Area El. 178'	116.0	1.24	4.05	0.1066	0.0781	Perforation of fan casing may occur. Due to the orientation of the fans, the postulated fan blade missile will not damage any safe shutdown equipment in the room. Therefore, protection is not needed.
Fan Blade	Control Area Exhaust Fan 1A,B-V402 (Centrifugal Fan)	Aux Bldg Control Area El. 155'	105.0	0.969	0.614	0.034	0.0781	Casing perforation will not occur; however, fan blade may exit through the flexible connector on the fan discharge. There is no safe shut-shutdown equipment in the room.

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TABLE 3.5-13

INTERNALLY GENERATED ROTATING COMPONENT
MISSILES OUTSIDE CONTAINMENT

MISSILE IDENTIFICATION	SOURCE OF MISSILE	LOCATION	MISSILE CHARACTERISTICS VELOCITY (FT/S)	DIA. (IN.)	WEIGHT (LBs)	CALCULATED MAX. STEEL PERF. DEPTH (IN.)	CASING THICKNESS	REMARKS
Fan Blade	FRVS Recir. Fan 1A thru F-V213 (Centrifugal Fan)	Reactor Bldg El. 132', 162' and 178'	248.0	1.4	5.42	0.318	0.1406	Perforation of the fan casing or flexible connector may occur. However, due to the orientation of the fans, only ceiling and floor may be hit. The calculated depth of the fan blade penetration on the concrete is 3.61". Since there are no safe shutdown commodities impacted, protection is not needed.
Fan Blade	FRVS Vent Fan 1A,B-V206 (Centrifugal Fan)	Reactor Bldg El. 145'	144.0	1.02	1.99	0.108	0.1406	Casing perforation will not occur; however, fan blade may exit through the flexible connector on the fan discharge. The calculated depth of the fan blade penetration into the concrete is 1.138". Due to the orientation of the fan, only the ceiling and floor could be hit. Therefore, protection is not needed.
Fan Blade	Control Room Emerg. Filter Fan 1A,B-V400 (Centrifugal Fan)	Aux Bldg Control Area El. 155'	197	0.772	0.764	0.115	0.1406	Casing perforation will not occur; however, fan blade may exit through the flexible connector on the fan discharge. The calculated depth of the blade penetration into the concrete is 1.09". There is no safe shutdown equipment in the room. Therefore protection is not needed.

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TABLE 3.5-13

INTERNALLY GENERATED ROTATING COMPONENT
MISSILES OUTSIDE CONTAINMENT

MISSILE IDENTIFICATION	SOURCE OF MISSILE	LOCATION	MISSILE CHARACTERISTICS VELOCITY (FT/S)	DIA. (IN.)	WEIGHT (LBs)	CALCULATED MAX. STEEL PERF. DEPTH (IN.)	CASING THICKNESS	REMARKS
Fan Blade	Battery Room Exhaust Fan 1A thru D-V406 (Centrifugal Fan)	Aux Bldg SDG Area El. 163'	81	0.846	0.23	0.014	0.0625	Casing perforation will not occur; however, fan blade may exit through the flexible connector on the fan discharge. The calculated depth of the fan blade penetration in the concrete is 0.086". Due to orientation of the fan, safe shutdown equipment will not be impacted and protection is not needed.
Fan Blade	Control Area Battery Exhaust Fan 1A,B-V410 (Centrifugal Fan)	Aux Bldg Control Area El. 178'	143	0.834	0.206	0.029	0.0625	Casing perforation will not occur; however, fan blade may exit through the flexible connector on the fan discharge. There are conduits that belong to A, C, and D channels in the room that may be needed for safe shutdown. However, the conduits are thicker than the calculated maximum steel perforation depth (0.029"), therefore, protection is not needed.
Fan Blade	Battery Room Exhaust Fan 1A,B-V416 (Centrifugal Fan)	Aux Bldg SDG Area El. 178'	81	0.846	0.23	0.014	0.0625	Casing perforation will not occur; however, fan blade may exit through the flexible connector on the fan discharge. There are conduits that belong to A, C, and D channels in the room that may be needed for safe shutdown. However, the conduits are thicker than the calculated maximum steel perforation depth (0.014"), therefore, protection is not needed.

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TABLE 3.5-13

INTERNALLY GENERATED ROTATING COMPONENT
MISSILES OUTSIDE CONTAINMENT

MISSILE IDENTI-FICATION	SOURCE OF MISSILE	LOCATION	MISSILE CHARACTERISTICS VELOCITY (FT/S)	DIA. (IN.)	WEIGHT (LBs)	CALCULATED MAX. STEEL PERF. DEPTH (IN.)	CASING THICKNESS	REMARKS
Fan Blade	Aux Bldg Battery Exhaust Fan 1A, B-V417 (Centri-fugal Fan)	Aux Bldg SDG Area El. 178'	78.5	0.984	0.792	0.027	0.0781	Casing perforation will not occur; however, fan blade may exit through the flexible connector on the fan discharge. There are conduits that belong to A, C, and D channels in the room that may be needed for safe shutdown. However, the conduits are thicker than the calculated maximum steel perforation depth (0.027"), therefore, protection is not needed.
Fan Blade	Control Equipment Supply Fan 1A, B-VH-407 (Centri-fugal Fan)	Aux Bldg SDG Area El. 178'	235	1.68	8.8	0.341	0.25	Perforation of fan casing may occur; however, the fan is inside a filter housing that is 3/16" thick. The calculated steel perforation after the fan blade penetration through the fan casing is 0.176". Therefore, the fan blade will not exit from the filter housing.
Fan Blade	Diesel Generator Panel Supply Unit Fan 1A, B-VH-408 (Centri-fugal Fan)	Aux Bldg SDG Area El. 163'	149	1.37	3.16	0.115	0.1875 (filter housing thickness)	Filter housing perforation will not occur.

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TABLE 3.5-13

INTERNALLY GENERATED ROTATING COMPONENT
MISSILES OUTSIDE CONTAINMENT

MISSILE IDENTI-FICATION	SOURCE OF MISSILE	LOCATION	MISSILE CHARACTERISTICS VELOCITY (FT/S)	DIA. (IN.)	WEIGHT (LBs)	CALCULATED MAX. STEEL PERF. DEPTH (IN.)	CASING THICKNESS	REMARKS
Fan Blade	Switchgear Room Unit Coolers 1A,B-VH-401 (Centri-fugal Fan)	Aux Bldg SDG Area El. 163'	157	3.31	8.09	0.094	0.1875 (filter housing thickness)	Filter housing perforation will not occur.
Fan Blade	Control Room Supply Unit 1A,B-VH-403 (Centri-fugal Fan)	Aux Bldg SDG Area El. 178'	174	1.45	4.867	0.178	0.1875	Casing perforation will not occur. Also, the fan is inside a filter housing.
Fan Blade	Control Area Smoke Vent Fan 10-V408 (Vane-Axial Fan)	Aux Bldg Control Area El. 178'	210	1.37	0.753	0.069	0.1875	Casing perforation will not occur. However, the fan blade may exit through the suction side flexible connector. There is no safe shut-down equipment within the room. Therefore, protection is not needed.
Fan Blade	Diesel Area Exhaust Fan 1A,B-V411 (Vane-Axial Fan)	Aux Bldg SDG Area El. 178'	281	1.72	0.902	0.092	0.1875	Casing perforation will not occur. However, the fan blade could exit through the suction side flexible connector. A 1/4" thick steel barrier is provided to enclose the section flexible connector.

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TABLE 3.5-13

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INTERNALLY GENERATED ROTATING COMPONENT
MISSILES OUTSIDE CONTAINMENT

MISSILE IDENTI-FICATION	SOURCE OF MISSILE	LOCATION	MISSILE CHARACTERISTICS VELOCITY (FT/S)	DIA. (IN.)	WEIGHT (LBs)	CALCULATED MAX. STEEL PERF. DEPTH (IN.)	CASING THICKNESS	REMARKS
Fan Blade	Diesel Generator Room Recir. Fan 1A thru H-V412 (Vane-Axial Fan)	Aux Bldg SDG Area El. 77'	260	3.33	23.9	0.383	0.25	Fan blade will penetrate through the fan casing. However, there are no safe shutdown equipment in the room. Therefore protection is not needed.
Fan Blade	Control Room Return Air Fan 1A,B-VH-415 (Vane-Axial Fan)	Aux Bldg Control Area El. 155'	362	1.26	0.72	0.151	0.1719	Casing perforation will not occur. However, the fan blade may exit through the suction flexible connector. There are no safe shutdown equipment in the room. Therefore, protection is not needed.
Fan Blade	RCIC Room Coolers 1A,B-VH-208 (Vane-Axial Fan)	Reactor Bldg El. 54'	205	1.36	0.758	0.0684	0.1875	Casing perforation will not occur. There is a wire screen on the suction of the Fan Cooler which will prevent a fan blade from leaving the cooler at an oblique angle.

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TABLE 3.5-13

INTERNALLY GENERATED ROTATING COMPONENT MISSILES OUTSIDE CONTAINMENT

MISSILE IDENTIFICATION	SOURCE OF MISSILE	LOCATION	MISSILE CHARACTERISTICS VELOCITY (FT/S)	DIA. (IN.)	WEIGHT (LBs)	CALCULATED MAX. STEEL PERF. DEPTH (IN.)	CASING THICKNESS	REMARKS
Fan Blade	RHR Room Coolers 1A thru H-VH-210 (Vane-Axial Fan)	Reactor Bldg El. 54'	281	2.12	4.59	0.220	0.25	Casing perforation will not occur. There is a wire screen on the suction of the fan cooler which will prevent a fan blade from leaving the cooler at an oblique angle.
Fan Blade	SACS Room Coolers 1A thru D-VH-214 (Vane-Axial Fan)	Reactor Bldg El. 102'	215	1.46	1.05	0.084	0.1875	Casing perforation will not occur. There is a wire screen on the suction of the fan cooler which will prevent a fan blade from leaving the cooler at an oblique angle.
Fan Blade	Core Spray Pump Room Coolers 1A thru H-VH-211 (Vane-Axial Fan)	Reactor Bldg El. 54'	230	1.61	1.598	0.11	0.1875	Casing perforation will not occur. There is a wire screen on the suction of the fan cooler which will prevent a fan blade from leaving the cooler at an oblique angle.

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TABLE 3.5-13

8

INTERNALLY GENERATED ROTATING COMPONENT
MISSILES OUTSIDE CONTAINMENT

MISSILE IDENTI-FICATION	SOURCE OF MISSILE	LOCATION	MISSILE CHARACTERISTICS VELOCITY (FT/S)	DIA. (IN.)	WEIGHT (LBs)	CALCULATED MAX. STEEL PERF. DEPTH (IN.)	CASING THICKNESS	REMARKS
Fan Blade	Intake Structure Supply Fan 1A thru D-V503 (Vane-Axial Fan)	Intake Structure El. 122'	250	2.72	8.49	0.22	0.25	Casing perforation will not occur. There is a wire screen on the suction of the fan cooler which will prevent a fan blade from leaving the cooler at an oblique angle.
Fan Blade	Intake Structure Exhaust Fan 1A thru D-V504 (Vane-Axial Fan)	Intake Structure El. 122'	250	2.72	8.49	0.22	.025	Casing perforation will not occur. There is no flexible connector on the suction or the discharge side.
Fan Blade	Traveling Screen Motor Room Fan OA,B-V558 (Vane-Axial Fan)	Intake Structure	138	1.368	0.746	0.04	0.1875	Casing perforation will not occur. The intake damper and vane guide on the suction of the fan prevents a fan blade from exiting in that direction and the vane guide on the discharge of the fan prevents a fan blade from leaving the fan housing on the discharge direction. Therefore, protection is not needed.

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TABLE 3.5-13

INTERNALLY GENERATED ROTATING COMPONENT
MISSILES OUTSIDE CONTAINMENT

9

MISSILE IDENTI-FICATION	SOURCE OF MISSILE	LOCATION	MISSILE CHARACTERISTICS			CALCULATED MAX. STEEL PERF. DEPTH (IN.)	CASING THICKNESS	REMARKS
			VELOCITY (FT/S)	DIA. (IN.)	WEIGHT (LBs)			
Impeller	SACS Pumps	Reactor Bldg El. 102'	98.8	16.1	1016.	0.267	0.625	No casing perforation.
Impeller	Fuel Pool Cooling Pump	Reactor Bldg El. 162'	121.6	5.3	46.4	0.136	0.59	No casing perforation.
Impeller	ECCS Jockey Pump	Reactor Bldg El. 54'	93.0	2.56	8.35	0.0629	0.43	No casing perforation.
Impeller	Torus Water Cleanup Pump	Reactor Bldg El. 54'	119.9	5.3	44.6	0.132	0.59	No casing perforation.
Impeller	Chilled Water Pump	Aux Bldg Control Area El. 155'	82.8	5.97	79.75	0.104	0.63	No casing perforation.
Impeller	D/G 1E Panel Chilled Water Pump	Aux Bldg Diesel Area El. 178'	94.5	3.04	11.79	0.068	0.39	No casing perforation.
Impeller	RACS Pump	Reactor Bldg El. 77'	79.6	6.24	83.66	0.0976	0.77	No casing perforation.

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TABLE 3.5-13

INTERNALLY GENERATED ROTATING COMPONENT
MISSILES OUTSIDE CONTAINMENT

MISSILE IDENTI-FICATION	SOURCE OF MISSILE	LOCATION	MISSILE CHARACTERISTICS VELOCITY (FT/S)	DIA. (IN.)	WEIGHT (LBs)	CALCULATED MAX. STEEL PERF. DEPTH (IN.)	CASING THICKNESS	REMARKS
Impeller	Service Water Booster Pump	Intake Structure El. 79'-8"	67.3	3.78	26.3	0.0596	0.51	No casing perforation.
Impeller	Service Water Pump	Intake Structure El. 93'	97.6	15.2	1215.5	0.314	0.75	No casing perforation.
Impeller	RWCU Recir. Pump	Reactor Bldg El. 132'	158.1	4.79	48.2	0.219	1.125	No casing perforation.
Impeller	RWCU Precoat Pump	Reactor Bldg El. 145'	62.4	4.31	31.8	0.053	0.5	No casing perforation.
Impeller	RWCU Holdup Pump	Reactor Bldg El. 145'	57.6	4.09	25.6	0.0439	0.801	No casing perforation.
Impeller	RWCU Backwash Pump	Reactor Bldg El. 132'	70.4	4.04	15.9	0.0423	0.43	No casing perforation.
Impeller	CRD Pump	Reactor Bldg El. 77'	120.6	3.91	21.4	0.109	0.675	No casing perforation.
Impeller	Service Water Dewater Pump	Reactor Bldg El. 54'	64.1	3.98	13.94	0.0347	0.5	No casing perforation

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TABLE 3.5-13

11

INTERNALLY GENERATED ROTATING COMPONENT
MISSILES OUTSIDE CONTAINMENT

MISSILE IDENTI-FICATION	SOURCE OF MISSILE	LOCATION	MISSILE CHARACTERISTICS VELOCITY (FT/S)	DIA. (IN.)	WEIGHT (LBs)	CALCULATED MAX. STEEL PERF. DEPTH (IN.)	CASING THICKNESS	REMARKS
Impeller	RCIC Pump	Reactor Bldg El. 54'	168.8	4.04	29.3	0.204	0.5	No casing perforation.
Impeller	HPCI Booster Pump	Reactor Bldg El. 54'	169.9	3.77	55.82	0.339	0.625	No casing perforation.
Impeller	HPCI Main Pump	Reactor Bldg El. 54'	224	2.65	18.2	0.330	0.687	No casing perforation.
(Later)	HPCI Turbine	Reactor Bldg El. 54'	(Later)	(Later)	(Later)	(Later)	(Later)	(Later)
(Later)	RCIC Turbine	Reactor Bldg El. 54'	(Later)	(Later)	(Later)	(Later)	(Later)	(Later)
(Later)	RPS MG Sets	Aux. Bldg Control Area El. 54'	(Later)	(Later)	(Later)	(Later)	(Later)	(Later)

DSER OPEN ITEM 29

QUESTION 410.11 (SECTION 3.5.1)

The FSAR states that fans are not considered as credible missile sources. Recently (Palo Verde, 1982) a fan at a nuclear facility generated a missile which penetrated the fan housing and damaged a safety-related structure. Provide a discussion of the effects of fan blades as a missile source and the means used to prevent damage of safety-related equipment for each fan.

RESPONSE

Insert → Delete

~~As discussed in the HCGS response to Question 410.12, we do not consider through-fan-housing missiles that would damage safety-related structures to be credible. The condition that existed at Palo Verde involved workmanship deficiencies as the blade locknut torque and blade tip angle did not meet the supplier's specification. As a result, the blade experienced fatigue failure and was ultimately propelled out of the fan housing at an angle that penetrated the flexible connections of the fan and impinged the containment liner plate. HCGS has conducted a survey of vane-axial and centrifugal fans in safety-related areas employing flexible connectors. We identified one instance where a postulated missile through the flexible connection of a vane-axial fan may have the potential to damage safe-shutdown equipment in the room. In order to prevent the postulated missile from damaging safe-shutdown equipment, a missile shield has been added to the design to withstand the impact of the postulated missile.~~

INSERT

Section 3.5 has been revised to provide the results of an analysis which shows that internally generated rotating component missiles have no adverse effect on plant safe shutdown capability.

DSER OPEN ITEM 29

QUESTION 410.12 (SECTION 3.5.1)

The FSAR states that rotating equipment which is not specifically identified does not constitute a missile hazard because of the "unliklihood" that a missile would penetrate the casing. Provide the results of a quantitative analysis to verify this conclusion.

RESPONSE

INSERT →

The possibility that any pump or fan other than those identified in Section 3.5.1, will fail at HCGS and generate a missile which has sufficient energy to penetrate a component casing is remote. A review of the analyses of internally generated missiles performed for Palo Verde verified that postulated missiles from pumps and fans (e.g., a pump impeller or fan blade) typically do not have sufficient energy to penetrate the component casings. The formulae used by Palo Verde to predict the penetration resulting from missile impact are provided in Reference 3.5-4.

Since HCGS uses pumps and fans which are designed and constructed in accordance with the same recognized industry codes and standards as those installed at Palo Verde, results of the rigorous analyses conducted for Palo Verde are indicative of the structural integrity of the HCGS equipment.

Delete

INSERT

Section 3.5 has been revised to provide the results of an analysis which shows that internally generated rotating component missiles have no adverse effect on plant safe shutdown capability.

QUESTION 410.13 (SECTION 3.5.1)

Provide a discussion of an analysis for each rotating component which verifies that the casing would be capable of retaining an internally generated missile. For each rotating component whose casing cannot retain the internally generated missile, verify that no secondary missiles will be generated from any internally generated missile.

RESPONSE

Insert →

~~The basis for considering it unlikely for rotating components,~~ other than those identified in Section 3.5.1, to break through their casings and adversely impact safety-related equipment are the following:

delete

1. A review of event reports on file at the Nuclear Safety Information Center; Oak Ridge National Loboratory, concerning failures of fans and missile generation indicated that no fan failures have resulted in generation of through-casing missiles in safety-related areas of a nuclear facility.

2. Small pump failures resulting in generation of missiles are considered less probable than fan failures resulting in generation of missiles because pump casings are generally thicker than fan casings and pump speeds are generally slower than fan speeds.

3. Even in the unlikely event that a rotating component does break through its casing, much of the missile's kinetic energy would be dissipated in moving through the casing; thereby decreasing the probability of the missile damaging a safety-related component. Therefore, generation of secondary missiles from the internally generated missiles described above is not considered credible.

4. It is even a lower probability that a rotating component would adversely affect redundant safety-related systems because redundant equipment is generally located in different areas or separated by barriers.

5. A review of a detailed analysis of internally generated missiles performed by Palo Verde verified that postulated missiles from pumps and fans (e.g. a pump impeller or fan blade) typically do not have sufficient energy to penetrate the component casing. Because Hope Creek uses pumps and fans that are designed and constructed in accordance with the same recognized industry codes and standards as those ~~installed at Palo Verde, the results of their rigorous~~

~~structural analysis is indicative of the integrity of HCGS equipment.~~

INSERT

Section 3.5 has been revised to provide the results of an analysis which shows that internally generated rotating component missiles have no adverse effect on plant safe shutdown capability.

DSER Open Item 31 (Section 3.5.1.3)

TURBINE MISSILES

The staff considers the turbine missile issue as an open item until the applicant agrees to:

(1) submit for NRC approval, within three years of obtaining an operating license, a turbine system maintenance program based on the manufacturer's calculations of missile generation probabilities, or

(2) volumetrically inspect all low pressure turbine rotors at the second refueling outage and every other (alternate) refueling outage thereafter until a maintenance program is approved by the staff; and conduct turbine steam valve maintenance (following initiation of power output) in accordance with present NRC recommendations as stated in SRP Section 10.2 of NUREG-0800.

RESPONSE

HCGS will submit for NRC approval within three years of obtaining an operating license, A Turbine System Maintenance Program based on the manufacturer's calculations of missile generation probabilities.

This response assumes that by that time, the NRC will have approved the manufacturer's calculation methodology which has already been submitted to the NRC for approval.

DSER Open Item No. 34 (Section 3.6.2)

UNRESTRAINED WHIPPING PIPE INSIDE CONTAINMENT

For high energy piping within the containment penetration area where breaks are not postulated, SRP Section 3.6.2 sets forth certain criteria for the analysis and subsequent augmented inservice inspection requirements. Breaks need not be postulated in those portions of piping within the containment penetration region that meet the requirements of the ASME Code, Section III, Subarticle NE-1120 and the additional requirements outlined in Branch Technical Position MEB 3-1 of SRP Section 3.6.2. Augmented inservice inspection is required for those portions of piping within the break exclusion region.

RESPONSE

For the information requested above, see the response to Question 210.14.

DSER Open Item No. 90 (Section 3.9.2.1)

VIBRATION MONITORING PROGRAM DURING TESTING

Piping vibration, thermal expansion, and dynamic effects testing will be conducted during a preoperational testing program. The purpose of these tests is to assure that the piping vibrations are within acceptable limits and that the piping system can expand thermally in a manner consistent with the design intent. During the Hope Creek plant's preoperational and startup testing program, the applicant will test various piping systems for abnormal, steady-state or transient vibration and for restraint of thermal growth. Systems to be monitored will include (1) ASME Code Class 1, 2 and 3 piping systems, (2) high energy piping systems inside seismic Category I structures, (3) high energy portions of systems whose failure could reduce the functioning of seismic Category I plant features to an unacceptable safety level, and, (4) seismic Category I portions of moderate energy piping systems located outside containment. Steady-state vibration, whether flow-induced or caused by nearby vibrating machinery, could cause 10^8 or 10^9 cycles of stress in the pipe during its 40-year life. For this reason, the staff requires that the stresses associated with steady-state vibration be minimized and limited to acceptable levels. The test program will consist of a mixture of instrumented measurements and visual observations by qualified personnel.

Additional information of the criteria to be used for determining acceptability of observed or measured vibration levers for NSSS piping systems need to be included in the FSAR.

RESPONSE

For the information requested above, see responses to Questions 210.29 and 210.30.

DSER Open Items 105, 106 (Section 4.2)

PLANT-SPECIFIC MECHANICAL FRACTURING ANALYSIS

APPLICABILITY OF SEISMIC AND LOCA LOADING EVALUATION

1. The mechanical fracturing analysis is usually done as a part of the seismic and LOCA loading analysis (see Item (2)). The staff has reviewed and approved the generic analytical method used by GE (described in NEDE-21175-3) to determine that fuel-rod mechanical fracturing will not occur as a result of combined seismic and LOCA loadings. However, the applicant has not demonstrated that this generic report is applicable to Hope Creek or presented an acceptable alternative. In either case, we require a plant specific analysis.

2. Earthquakes and postulated pipe breaks in the reactor coolant system would result in external forces on the fuel assembly. SRP Section 4.2 and associated Appendix A state that fuel assembly coolability should be maintained and that damage should not be so severe as to prevent control rod insertion when required during these low probability accidents. The SRP recommends acceptance criteria to achieve these objectives.

   The entire seismic and LOCA loading evaluation has been described by GE in the approved topical report NEDE-21175-3.

   This item is similar to Item 1. The applicant must demonstrate that NEDE-21175-3 is applicable to Hope Creek or provide an acceptable alternative along with a plant-specific analysis to show that the criteria given in SRP Section 4.2 Appendix A, are met.

RESPONSE

In accordance with the methods described in NEDE-21175-3 (LTR), the HCGS fuel design was analyzed for the plant unique seismic and annulus pressurization (AP) loadings. However, the seismic and AP loadings for Hope Creek were calculated by a linear dynamic analysis using the HCGS reactor building model with GE's detailed RPV model.

To address the fuel lift, a screening assessment was performed comparing the Hope Creek unique combined (seismic and AP) input loads at the top of the RPV support skirt (the

DSER Open Items 105, 106 (Section 4.2) (Cont'd)

load input point to the LTR model) with the input loads of other similar BWR plants for which plant-unique nonlinear LTR analyses were performed.

The screening assessment showed that the HCGS plant-unique input loads are well below the input loads of the comparison plants. Since the nonlinear-analysis fuel lift values for these plants were well below the acceptable fuel-design limits, the HCGS fuel-lift values are expected to be negligible.

DSER Open Item No. 113 (Section 5.3.4)

GE PROCEDURE APPLICABILITY

To demonstrate that the GE Procedure Y 1006A006 is applicable to Hitachi fabricated vessel, provide:

(a) GE Procedure Y 1006A006

(b) Test results and analysis of Hitachi fabricated materials and the supplier which show the GE Procedure will conservatively predict the RTNDT for the Hitachi forgings, plates, and welds.

The plate/forge materials, which form the data base for the analysis, must be melted, cross-rolled or forged, and heat treated to a condition equivalent to that of the Hitachi plate/forge material.

The weld materials, which form the data base for the analysis, must be fabricated using equivalent wire flux and heat treatment as the Hitachi weld materials.

RESPONSE

For the information requested above, see the response to Questions 251.2.

QUESTION 251.2

To demonstrate that the GE Procedure Y 1006A006 is applicable to Hitachi fabricated vessel, provide:

a. GE Procedure Y 1006A006

b. Test results and analysis of Hitachi fabricated materials and its supplier which shows the GE Procedure will conservatively predict the RT_{NDT} for the Hitachi forgings, plates, and welds.

The plate/forge materials which forms the data base for the analysis, must be melted, cross-rolled or forged and heat treated to an equivalent condition as the Hitachi plate/forge material.

The weld material, which form the data base for the analysis must be fabricated using equivalent wire flux and heat treatment as the Hitachi weld materials.

RESPONSE

The applicability of General Electric Procedure Y 1006A006, revision 1 (attached) to the Hitachi-fabricated Hope Creek Unit 1 reactor pressure vessel (RPV) is demonstrated by Tables 251.2-1 and 251.2-2. These tables compare the chemistries, heat treatments, and mechanical properties of the materials that form the data base for the application of Y1006A006 with the properties of the HCGS RPV materials. Table 251.2-1 provides data for plate materials, and Table 251.2-2 provides data for forgings. The comparisons indicate that for both plates and forgings there are no significant differences in these properties between the Y1006A006 materials and the HCGS RPV materials.

Further evidence of the compatibility of the HCGS RPV material is presented in Tables 251.2-3 and 251.2-4, which compare Charpy V-notch test results. As shown in Table 251.2-3, the plates fabricated by Japan Steel/Hitachi have toughness properties equivalent to the Y1006A006 data-base materials, although they were evaluated at test temperatures 10°F lower. Similarly, as shown in Table 251.2-4, the Japan Steel/Hitachi forgings demonstrate a -10°F notch toughness comparable to results for the Y1006A006 forgings, which were tested at +50°F.

Evidence of the equivalence of the Y1006A006 and Hitachi weld materials is given in Table 251.2-5, which compares their respective chemistries, tensile properties, and thermal treatments. Except for the Ni content, these materials are very similar, although the Hitachi weld metals are generally lower in phosphorus and sulfur content.

Table 251.2-6 compares the Charpy V-notch impact-test results for Y1006A006 and Hitachi weld materials. The Hitachi materials correspond well with the notch toughness values for the Y1006A006 materials and, in fact, are generally superior. The submerged-arc weld materials used for

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fabrication of the HCGS RPV are not presented in this response because their toughness properties are suitable to meet the requirements of Appendix G of 10 CFR 50 for establishing reference temperatures, and it was not necessary to apply procedure Y1006A006.

Table 251.2-1

Comparison of SA 533 Plate Material
Used as the Data Base for GE Procedure Y1006A006 Versus SA533 Material Manufactured by Japan Steel Works for Hope Creek Unit 1 Reactor Pressure Vessel

Grade	Thickness (in.)	Source	No.[1]	Average Composition of Materials (wt%) C	Mn	P	S	Si	Ni	Cr	Mo	Heat Treatment	Orient.	Yield Strength (KSI)	Ultimate Tensile Strength (Ksi)	Percent % El.
A533	6-6.5	GE	5	0.21	1.32	0.009	0.014	0.18	0.51	—	0.48	1625F-6Hr.-Agitated Brine-Q+1200F-6Hr.-Brine-Q+1125F-30Hr.-FC to 600F	Long. Tran.	69.2 66.0	90.4 88.4	27.9 26.6
A533	7-7.5	Comb.	6	0.22	1.36	0.011	0.014	0.19	0.53	—	0.49	1675F-4Hr.-AC+1600F-4Hr.-Agitated WQ+1225F-4Hr.-FC+1150F-40Hr.-FC	—	—	—	—
A533	8-8.5	GE	4	0.22	1.39	0.011	0.018	0.20	0.54	0.11	0.49	1775F-8.5Hr.-Agitated Brine-Q+1200F-6Hr.-Brine-Q+1125F-30Hr.-FC	—	—	—	—
A533	8.5-9	Comb.	1	0.22	1.38	0.011	0.013	0.21	0.44	—	0.49	1675F-4Hr.-AC+1600F-4Hr.-Agitated WQ+1225F-4Hr.-FC+1150F-40Hr.-FC	Tran.	68.3	88.6	25.4
A533	9.5-10	West.	6	0.21	1.31	0.011	0.017	0.22	0.57	0.14	0.47	1600F-4Hr.-Agitated WQ-1225F-4Hr.-AC+1150F-40Hr.-FC	Tran. Long.	66.4 66.2	86.3 87.4	24.3 26.0
A533	11.5-12	Comb.	3	0.23	1.31	0.010	0.015	0.19	0.55	—	0.58	1675F-4Hr.-AC+1600F-4Hr.-Agitated WQ+1225F-4Hr.-FC+1150F-40Hr.-FC	Tran.	64.4	86.7	26.5
A533	11.5-12	West.	4	0.21	1.35	0.013	0.022	0.24	0.51	—	0.48	1600F-4Hr.-Agitated WQ+1225F-4Hr.-AC+1150F-27Hr.-FC	Long. Tran.	66.7 67.8	87.3 86.9	26.2 26.0
A533[2]	6-6.5	Japan Steel		0.20	1.45	0.012	0.008	0.31	0.63	—	0.56	(1580F-1634F)-3.4Hr.-Q+(1202F-1238F)-3.3Hr.+(1112F-1130F)-40.5Hr.	—	70.2	92.5	26.5
A533	6-6.5	Japan Steel		0.20	1.43	0.010	0.008	0.30	0.56	—	0.54	"	—	70.8	92.3	25.1
A533	6-6.5	Japan Steel		0.22	1.43	0.009	0.008	0.29	0.58	—	0.59	"	—	69.3	91.8	25.0
A533	6-6.5	Japan Steel		0.19	1.44	0.010	0.012	0.30	0.56	—	0.50	"	—	62.7	87.5	27.5
A533	6-6.5	Japan Steel		0.20	1.46	0.010	0.011	0.27	0.54	—	0.51	"	—	66.2	85.4	24.6
A533	6-6.5	Japan Steel		0.018	1.49	0.008	0.010	0.31	0.57	—	0.50	"	—	68.6	90.5	25.6

[1] No. = Number of plates tested
[2] = SA 533, Gr.B, Cl.1

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Table 251.2-2

Comparison of SA 508 Forging Material
Used as the Data Base for GE Procedure Y1006A006 Versus SA508 Material Manufactured by Japan Steel Works for Hope Creek Unit 1 Reactor Pressure Vessel

de	Thickness (in.)	Source	No.	C	Mn	P	S	Si	Ni	Cr	Mo	V	Heat Treatment	Orient.	Yield Strength (Ksi)	Ultimate Tensile Strength (Ksi)	[illegible]
				Average Composition of Materials (wt %)													
08 Cl.2	8-8.5	West.	1	0.19	0.65	0.010	0.007	0.23	0.69	0.33	0.60	0.02	1550F-9Hr.-WQ+1210F-12Hr.-AC+1125F-11Hr.-FC	Tang.	72.1	91.3	69
08 Cl.2	9-9.5	West.	1	0.22	0.63	0.009	0.011	0.24	0.68	0.34	0.59	0.02	1585F-11Hr.-Double WQ-1220F-22Hr.-AC+1110F-6Hr.+50°/Hr. to 600F	Tang.	58.9	82.1	70
08 Cl.2	15-20	GE	1	0.21	0.60	0.010	0.007	0.24	0.67	0.33	0.58	0.04	1615F-9Hr.-Agitated WQ+1230F-20Hr.-WQ+1125F-30Hr.-100°/Hr. to 600F-AC	Tang.	60.0	82.1	73
08 Cl.2	20-25	Ladish	4	0.23	0.63	0.009	0.010	0.26	0.78	0.35	0.63	0.045	1650F-8Hr.-AC+1650F-8Hr.-WQ+1275F-24Hr.-WQ+1150F-30Hr.-FC to 600F-AC	Tang.	62.5	87.0	64
08 Cl.2	6.7	Japan Steel		0.16	0.72	0.010	0.009	0.32	0.84	0.39	0.62	—	(1634F-1643F)Austenitize-9.1Hr.+(1211F-1220F)Temper-16Hr.+1144F-PWHT-40Hr.	—	71.0	88.4	70
08 Cl.2	6.7	Japan Steel		0.15	0.70	0.011	0.011	0.32	0.81	0.38	0.63	Tr.	(1652-1670F)Austenitize-11Hr.+(1220-1230F)-Temper-16.5Hr.+1156F-PWHT-40Hr.	—	65.1	82.5	72

No. = Number of forgings tested

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Table 251.2-3

Comparison of Notch Toughness Information for Japan Steel and Y1006A006 Plate Material

Grade	Thickness (in.)	Source	Orientation	No.[1]	Test Temperature (°F)	Average Absorbed Energy (ft-lb)	Average Lateral Expansion (mils)
			1/4T Charpy V-Notch Test Results				
A533	6-6.5	CE	Transverse	5	+ 50	60	44
↓	7-7.5	Comb.	↓	6	↓	56	45
↓	8-8.5	CE	↓	4	↓	60	40
↓	8.5-9	Comb	↓	1	↓	53	40
↓	11.5-12	Comb.	↓	3	↓	47	36
↓	11.5-12	West.	↓	4	↓	44	40
SA533, Gr.B, Cl.1	6.2-6.8	Japan Steel	↓	See Below[2]	+ 40	44	34
↓	↓	↓	↓		↓	50	38
↓	↓	↓	↓		↓	81	57
↓	↓	↓	↓		↓	64	50
↓	↓	↓	↓		↓	54	40
↓	↓	↓	↓		↓	52	41

1 No = Number of plates tested

2 Each row of data represents a heat of material used in the beltline region of the Hope Creek Unit 1 RPV.

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Table 4b251.2-4

Comparison of Notch Toughness Information for Japan Steel and Y1006A006 Forgings

Grade	Thickness (in.)	Source	Orientation	No.[1]	1/4T Charpy V-Notch Test Results: Test Temperature (°F)	Average Absorbed Energy (ft-lb)	Average Lateral Expansio[n] (mils)
A508 Class 2	8-8.5	West.	Tang.	1	+ 50	81	60
↓	9-9.5	West.	Tang.	1	↓	96	64
↓	15-20	GE	Long.	1	↓	96	55
↓	20-25	Ladish	N.R.	4	↓	48	NR
ASME SA508, Class 2	6.7	Japan Steel/ Katsuta Works, Hitachi Ltd.	Long.	See Below[2]	- 10	80	66
ASME SA508, Class 2	6.7	Japan Steel/ Katsuta Works, Hitachi Ltd.	Long.		- 10	77	62

[1] No. = Number of forgings tested

[2] Each row of data represents a heat of material used in the fabrication of the low pressure core injection nozzles for Hope Creek Unit 1 RPV

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Table 251.2-5

Comparisons of Y1006A006 and Hitachi Shielded Metal Arc Weld Material

Heat/Lot	Chemical Composition (wt.%) C	Ni	Mn	Si	P	S	Mo	V	Cu	Yield Strength (ksi)	Ultimate Tensile Strength (ksi)	Reduction of Area (%)	Heat Treatment
Y1006A006 DATA BASE:													
402P3162/B[illegible]6B27AE	0.066	0.83	1.06	0.46	0.02	0.018	0.49	0.019	0.03	78.7	90.7	42.8	1150°F ± 20° for 50 hours
4[illegible]/B4[illegible]8B27AF	0.06	0.98	1.09	0.36	0.013	0.017	0.52	0.02	0.03	73.5	83.5	71.2	"
03[illegible]048/B525B27AF	0.04	0.96	1.23	0.40	0.014	0.014	0.53	0.02	0.09	78.0	91.0	64.7	"
L83978/J414B27AD	0.08	1.06	1.15	0.51	0.017	0.014	0.54	0.02	0.02	83.7	94.5	69.5	"
401C0371/B5[illegible]4B27AE	0.05	1.04	1.18	0.37	0.012	0.012	0.56	0.02	0.03	84.2	94.4	68.2	"
492LA071/A421B27AE	0.07	0.95	1.06	0.37	0.018	0.025	0.50	0.02	0.04	72.0	84.5	72.7	"
422K6311/G313A27AD	0.06	1.00	1.2[illegible]	0.31	0.016	0.013	0.54	0.02	0.01	81.3	91.5	74.5	"
640892[illegible]B27AE	0.08	1.00	1.20	0.44	0.015	0.018	0.55	0.02	0.39	76.5	90.0	71.0	"
[illegible]/B403B27AG	0.06	0.97	1.16	0.35	0.0[illegible]	0.021	0.51	0.02	0.04	68.0	80.5	71.4	"
HITACHI:													
51[illegible]-01205	0.072	0.54	1.20	0.42	0.016	0.011	0.45	—	0.09	[illegible]3.6	94.6	67.9	1112-1170°F 40 hours
519-01205	0.08	0.56	1.10	0.40	0.011	0.012	0.46	—	—	73.0	85.5	71.7	"
504-01205	0.07	0.48	1.01	0.42	0.011	0.008	0.44	—	—	69.8	83.3	68.2	"

DSER OPEN ITEM 113

Table 4251.2-6

Comparison of CVN Test Results for Y1006A006 and Hitachi Weld Materials

Source	Heat/Flux	Process	Test Temp (°F)	Absorbed Energy (ft.lb)	Lateral Expansion (mils)	Shear (%)
Y1006A006	03L048/B525B27AF	SMAW	0	61, 75, 79	44, 58, 59	50, 60, 60
		↓	+ 40	104, 108	75, 77	80, 80
		↓	+130	122, 123, 126	89, 83, 91	100, 100, 100
	02R486/J404B27AG	↓	- 10	52, 64, 66	39, 45, 46	40, 40, 40
		↓	+ 40	84, 87	63, 68	60, 60
		↓	+130	121, 124, 129	91, 96, 95	100, 100, 100
	L83978/J414B27AD	↓	- 20	51, 52, 81	37, 40, 63	35, 50, 40
		↓	+ 40	120, 123	72, 73	80, 80
		↓	+ 72	128, 140	78, 81	90, 90
	401B0371/B504B27AE	↓	0	80, 85, 82	63, 62, 60	35, 50, 35
		↓	+ 40	95, 97	71, 76	40, 75
		↓	+ 70	111, 107, 109	87, 85, 77	80, 90, 80
	402P3162/H426B27AE	↓	- 10	60, 54, 68	44, 37, 53	40, 30, 30
		↓	+ 40	96, 99	57, 68	60, 60
		↓	+212	119, 122, 124	93, 90, 68	100, 100, 100
	492LA871/A421B27AE	↓	0	50, 51, 57	36, 38, 40	30, 40, 45
		↓	+ 40	135, 137	84, 80	90, 80
	422K8511/G313A27AD	↓	- 20	65, 74, 127	44, 48, 76	40, 50, 60
		↓	+ 25	107, 108	74, 80	80, 70
	640892/J424B27AE	↓	0	55, 62, 62	38, 44, 48	35, 40, 40
		↓	+ 40	56, 75	42, 55	50, 60
		↓	+130	118, 122, 130	87, 89, 82	100, 100, 100
	401P2871/H430B27AE	↓	0	27, 50, 56	25, 42, 46	40, 45, 45
		↓	+ 10	75, 76, 107	60, 62, 74	60, 50, 80
		↓	+ 40	90, 100	71, 76	70, 80
	07R458/H403B27AG	↓	0	59, 61, 70	51, 52, 58	50, 50, 60
		↓	+ 40	99, 101	77, 78	80, 75
		↓	+ 72	106, 110	85, 87	80, 80

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351.2-6

Table 9A (continued)

Comparison of CVN Test Results for Y1006A696 and Hitachi Weld Materials

Source	Heat/Flux	Process	Test Temp (°F)	Absorbed Energy (ft-lb)	Lateral Expansion (mils)	Shear (%)
Hitachi	510-01205	SMAW	+ 10	90, 73, 48	70, 64, 38	60, 40, 30
				98, 87, 92	65, 66, 65	50, 50, 50
	519-01205	↓	+ 10	110, 110, 107	87, 78, 70	75, 75, 80
	504-01205	↓	+ 10	130, 120, 123	89, 84, 92	75, 80, 75

DSER OPEN ITEM 113

BIS IDENT: MTHDS RTNDT VESSEL STLS
NUCLEAR ENERGY BUSINESS OPERATIONS
GENERAL ELECTRIC

Y1006A006	SH NO. 1
REV 1	

REVISION STATUS SHEET

DOCUMENT TITLE METHODS FOR ESTABLISHING INITIAL REFERENCE TEMPERATURES (RT_{NDT}) FOR VESSEL STEELS FOR CERTAIN PLANTS

LEGEND OR DESCRIPTION OF GROUPS

TYPE DESIGN PROCEDURE

FMF

MPL ITEM NO. N/A

| - DENOTES CHANGE

REVISIONS C

1	SE CARTER S.E. Carter 3-28-??	RFF
	RETYPED WITH CHANGES PER NH19110 CHK BY: [signature] EA HARTMAN	

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PRINTS TO

MADE BY	APPROVALS	DEPT	LOCATION
JF COPELAND 12/19/78	RA PROEBSTLE 12/21/78	NTD	SAN JOSE
	REISSUED RF FRANCIOSE 1/25/79	CONT ON SHEET 2	SH NO. 1

DSER OPEN ITEM 113

1. SCOPE OF APPLICATIONS AND OBJECTIVES

1.1 This procedure describes the method to be used for establishing the initial reference temperature (RT_{NDT}) for ferritic vessel steels for older plants where fracture toughness data may be incomplete. These methods represent a General Electric alternate position to the NRC Regulation 10CFR50 Appendix G for these plants.

2. METHODS

2.1 Vessel Plate (SA-533 Gr. B Cl. 1):

Predicted limiting property - either NDT (Nil-Ductility Transition Temperature) or transverse CVN (Charpy V-Notch) 50 ft-lb T.T. (Transition Temperature)

Usual data available - NDT and/or longitudinal CVN at +10 or +40°F

RT_{NDT} prediction method -

Operate on lowest longitudinal CVN ft-lb to get at least 50 ft-lb T.T. by adding 2°F per ft-lb or by plotting a curve (ft-lb versus temperature), where possible. Add additional 30°F to convert from longitudinal to transverse 50 ft-lb T.T.

NOTE: Where transverse CVN impact data are available, but the 50 ft-lb T.T. is not met, operate on the lowest CVN ft-lb to get at least 50 ft-lb T.T. by adding 3°F per ft-lb or by plotting a curve (ft-lb vs temperature), where possible. This extrapolation is valid for CVN test temperatures only in the range (-25° to +50°F).

Derive NDT, where missing, as equal to longitudinal CVN 35 ft-lb T.T.

RT_{NDT} is higher of NDT or transverse CVN 50 ft-lb T.T. -60°F

2.2 Forings (SA-508 Cl. 2):

Predicted limiting property - NDT or transverse CVN 50 ft-lb T.T.

Usual data available - NDT and/or CVN at single temperature

RT_{NDT} prediction method -

Derive CVN 50 ft-lb T.T. as for plate.

When only CVN values are available, estimate NDT as the lower of +70°F or the CVN test temperature where at least 100 ft-lb or 50 percent shear is achieved.

RT_{NDT} is higher of NDT or transverse CVN 50 ft-lb T.T. -60°F.

2.3 Weld Metal (Used to Join SA-533 Gr. B CL. 1 Plates and SA-508 CL. 2 Forgings):

Predicted limiting property - CVN 50 ft-lb T.T.

Usual data available - CVN values at single or at several test temperatures.

RT_{NDT} prediction method -

Operate on lowest CVN ft-lb to get at least 50 ft-lb T.T. by adding 2°F per ft-lb or by plotting a curve (ft-lb versus temperature), where possible.

RT_{NDT} is the CVN 50 ft-lb T.T. - 60°F. If NDT is available, it will be considered also. In absence of NDT data, RT_{NDT} shall not be lower than -50°F.

2.4 Vessel Plate (SA-533 Gr. B Cl. 1) and Forging (SA-508 Cl. 2) Weld HAZ:

RT_{NDT} assumed same as for base material. Weld procedure qualification test requirements indicate this assumption is valid.

2.5 Bolting Material (SA-540 Gr. B24):

CVN 45 ft-lb and 25 MLE (Mils Lateral Expansion) are required at no higher than preload temperature or Lowest Service Temperature (LST)

Usual data available - CVN ft-lb and MLE at +10°F

LST prediction method -

If preceding CVN requirements are met at test temperature, then it is LST.

If at least 30 ft-lb, but less than 45 ft-lb and 25 MLE, are met at test temperature, then add 60°F to the test temperature for LST.

DSER Open Item No. 114 (Section 5.3.4)

COMPLIANCE WITH NB2360 OF THIS SUMMER 1972 ADDENDA OF THE 1971 ASME CODE

To demonstrate compliance with the qualification and calibration requirements of NB 2360 of the Summer 72 Addenda to the 1971 edition of the ASME Code, indicate the qualification and calibration program requirements that were used for the RCPB materials and indicate how these requirements satisfy the calibration and qualification requirements of NB 2360 of the Summer 72 Addenda to the ASME Code.

RESPONSE

For the information requested above, see the response to Question 251.3.

QUESTION 251.3

To demonstrate compliance with the qualification and calibration requirements of NB 2360 of the Summer 72 Addenda to the 1971 edition of the ASME Code, indicate the qualification and calibration program requirements, which were used for the RCPB materials and indicate how these requirements satisfy the calibration and qualification requirements of NB 2360 of the Summer 72 Addenda to the ASME Code.

RESPONSE

As indicated in Section 5A.3:

a. The main steam piping material was tested in accordance with the Summer, 1972 Addenda to the 1971 Edition of Section III of the ASME B&PV Code.

b. The flued-head fitting material was tested in accordance with the Winter, 1973 Addenda to the 1971 Edition of Section III of the ASME B&PV Code.

c. The safety/relief valves were exempted from testing because of their 6-inch size.

d. The main steam isolation valves were also exempted from testing at the time of purchase.

The reactor pressure vessel was procured to the Winter, 1969 Addenda to the 1968 Edition of Section III of the ASME B&PV Code. Information from GETSCO, Tokyo, indicates that Hitachi impact tested the RPV material in accordance with paragraph NB 2360 of the Summer, 1972 Addenda of the 1971 Edition of the ASME B&PV Code.

DSER OPEN ITEM 114

DSER Open Item No. 115 (Section 5.3.4)

DROP WEIGHT AND CHARPY V-NOTCH TESTS FOR CLOSURE FLANGE MATERIALS

Provide drop weight test and Charpy V-notch test results from the closure flange region materials to demonstrate compliance with the closure flange requirements of Appendix G, 10 CFR 50.

RESPONSE

For the information requested above, see the response to Question 251.4.

QUESTION 251.4:

Provide drop weight test and Charpy V-notch test results from the closure flange region materials to demonstrate compliance with the closure flange requirements of Appendix G, 10 CFR 50.

RESPONSE

Available drop-weight and Charpy V-notch test results for the Hope Creek Unit 1 closure flange materials are provided below:

Material	Orientation	NDT Temp. (°F)	Test Temp. (°F)	Absorbed Energy (Ft-lbs)	Lateral Expansion (Mils)
SA508, Cl.2 (Head Flange)	Longitudinal	-20/	-40	64.1,70.6,20.8,77.1	48,51,11,58,
		-10	-10	93.1,114.7,106.6,	64,78,62,55,
		@180°		87.8,97.1,71.9	64,49
		AWAY	10	81.1,108,133.6,	49,68,78,95,
				137.6,165.1	68,74
			40	157.4,121.5,137.6,	89,73,77,86,
				134.9,144.3,137.6	79,85
			60	199.9,154.8,159.9,	77,69,88,87,
				195.4,144.3,170.1	82,73
SA508,Cl.2 (Shell Flange)	Longitudinal	-10	10	120.1,122.8,130.9,	77,81,83,81,
				130.9,132.3,116.1	77,64
			-10	120.1,95.8,128.2,	72,58,80,75,
				109.3,101.2,87.8	59.57
			+40	141.6,134.9,141.6,	81,77,84,82,
				145.6,167.6,182.4	85,89
			-40	13.4,69.3,59.0,55.2,	7,48,41,38,
				74.5,101.2	54,68

DSER OPEN ITEM 115

DSER Open Item No. 116 (Section 5.3.4)

CHARPY V-NOTCH TESTS DATA FOR BASE MATERIALS AS USED IN SHELL COURSE NO. 3

Provide Charpy V-notch data and analysis from base materials that are similar to the base materials used in fabrication of shell course No. 3 to demonstrate that the upper shelf energy properties of the plates in shell course No. 3 exceed the requirements of Paragraph IV.A.1 of Appendix G, 10 CFR 50.

RESPONSE

For the information requested above, see the response to Question 251.5.

QUESTION 251.5

Provide Charpy V-notch data and analysis from base materials, which are similar to the base materials used in fabrication of shell course No. 3, to demonstrate that the upper shelf energy properties of the plates in shell course No. 3 exceed the requirements of Paragraph IV.A.1 of Appendix G, 10 CFR 50.

RESPONSE

Table 5A-1 provides drop-weight NDT information and Charpy V-notch test results for the materials from shell courses 4 and 5 as well as information for the materials from shell course No. 3. Table 5A-3 compares the heat treatments, the chemistries, and the mechanical properties of these shell course materials and demonstrates that the materials from shell courses 4 and 5 should be considered equivalent to those from shell course No. 3. This equivalence and the suitable upper-shelf energies for the plates from shell courses 4 and 5, as presented in Appendix 5A, demonstrate that plates from shell course No. 3 should be considered to have upper-shelf energies that meet or exceed the requirements of Apppendix G of 10 CFR Part 50.

DSER OPEN ITEM 116

DSER Open Item 127 (Section 6.2.1.6)

OPERABILITY TESTING OF VACUUM BREAKERS

Also, the vacuum breakers should be operability tested at monthly intervals to assure free movement of the valves. To minimize the potential for steam bypass, we will require the applicant to commit to (1) perform operational testing of the torus to drywell vacuum breakers once each month; and (2) perform a leakage test of the drywell to torus vent system at the end of each refueling outage. We will include these periodic tests in the technical specification.

RESPONSE

(1) H.C.O will commit to perform operational testing of the torus to drywell vacuum breakers at a frequency of once per 31 days. This requirement should be included in the HCGS Technical Specifications.

(2) A leakage test of the drywell to torus vent system will be performed at the same frequency as the containment type A test required by 10 CFR 50 Appendix J.

DSER Open Item No. 131 (Section 6.2.3)

ADMINISTRATION OF SECONDARY CONTAINMENT OPENINGS

The applicant has identified in FSAR Table 6.2-14 the openings into the reactor building enclosure. However, the applicant has not indicated whether the secondary containment openings are provided with position indicators and alarms with local and main control room readout and/or are under administrative controls to prevent their impacting the secondary containment functional integrity. We will require this information, and will report on this matter in a supplement to this SER.

RESPONSE

All openings listed in revised Table 6.2-14 and shown on Figures 6.2-20 thru 6.2-25 are alarmed and annunciated in the control room as indicated in revised Section 6.2.3.2.1.

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jeopardize the integrity of the reactor building. In addition, all the personal access doors and hatches to the reactor building are monitored and alarmed in the main control room.

6.2.3.2.2 Reactor Building Isolation System

The reactor building isolation system is described in Section 9.4.2.

6.2.3.2.3 Containment Bypass Leakage

Upon receipt of a LOCA or other high radioactivity signal, the FRVS is actuated automatically and simultaneously with reactor building isolation and shutdown of the normal RBVS. Radioactivity that exfiltrates the primary containment is collected and passed through the FRVS as described in Section 6.8.

Penetrations that pass through both primary and reactor building barriers and that have isolation valves, seals, gaskets, or welded joints are considered potential bypass leakage paths. Potential leakage paths that could bypass the areas serviced by the FRVS have been evaluated. Table 6.2-15 identifies those lines penetrating the primary containment that do not terminate inside the reactor building or in a closed system outside primary containment within the reactor building. Section 6.2.4.3.5 provides an evaluation of closed systems outside primary containment. Closed systems outside primary containment are considered effective bypass leakage barriers because they are dependable (i.e., Seismic Category I and Quality Group B) systems that are water filled by the use of the system jockey pumps. The systems are maintained leak tight by periodic visual inspection and the leak detection provisions identified in Section 5.2.5.2.2.

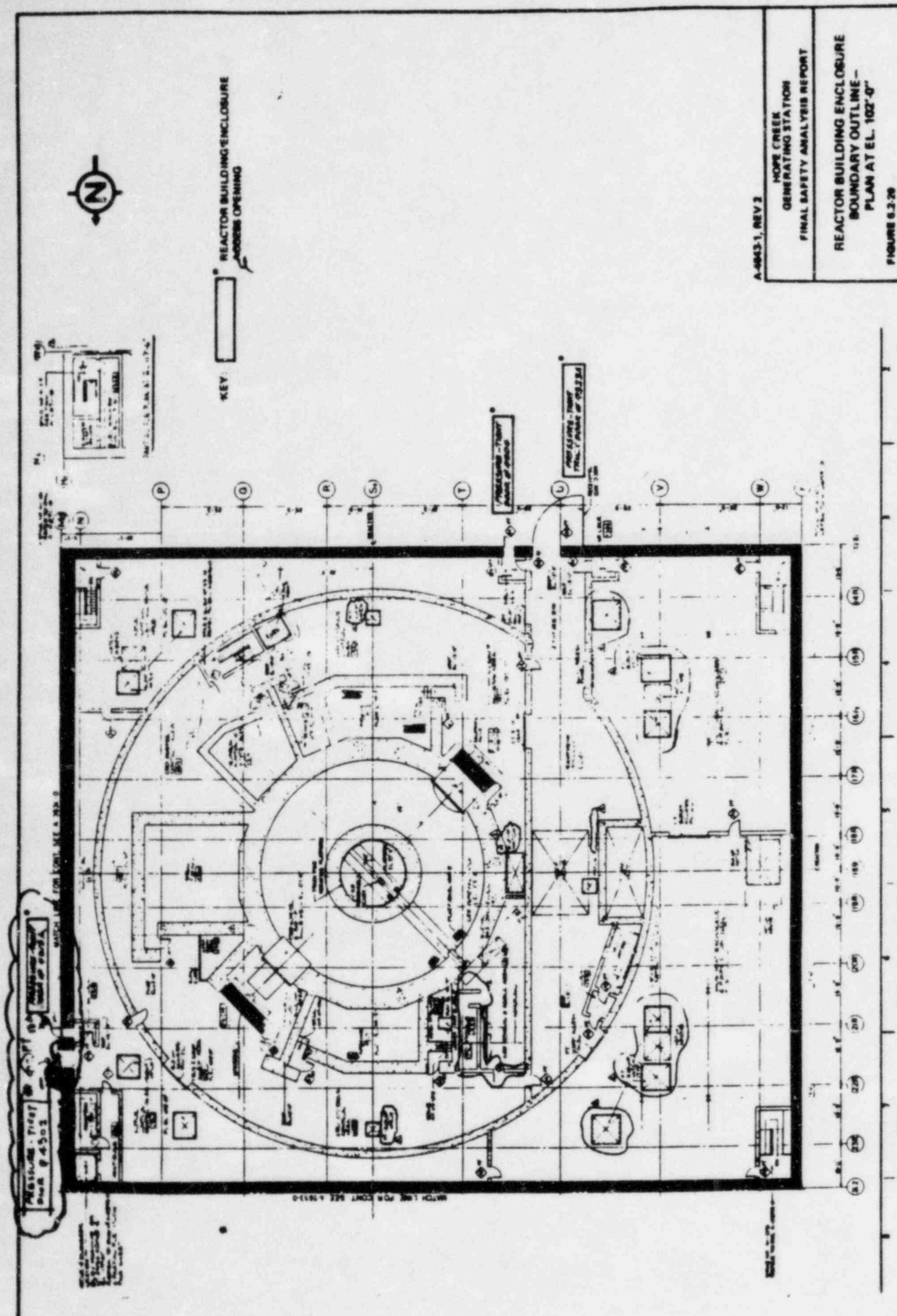
The types of bypass leakage barriers employed by lines listed in Table 6.2-15 are:

a. Redundant primary containment isolation valves

TABLE 6.2-14

OPENINGS IN REACTOR BUILDING ENCLOSURE

~~Access~~ Opening Number	Elev, ft	Near Column Coordinates	Type of ~~Access~~ Opening
~~S13~~	~~102-0~~	~~44.2, Ka~~	~~Pressure-tight door~~
4304	102-0	13.6, T	Pressure-tight door
4323A	102-0	13.6, U	Pressure-tight door
4313A	102-0	21R, Md	Pressure-tight door
	132-0	21R, V	Equipment hatch
	132-0	16R, V	Equipment hatch
	132-0	18.9, W	Equipment hatch
	132-0	15R, P	Equipment hatch
	132-0	18.9, V	Blowout panels
4501A	145-0	17R, P	Pressure-tight door
	145-0	18.9, V	Blowout panels
	162-0	18.9, V	Blowout panels
4302	102-0	22.R, Md	Pressure-tight door
4401A	132-0	19.R, Q	Pressure-tight door



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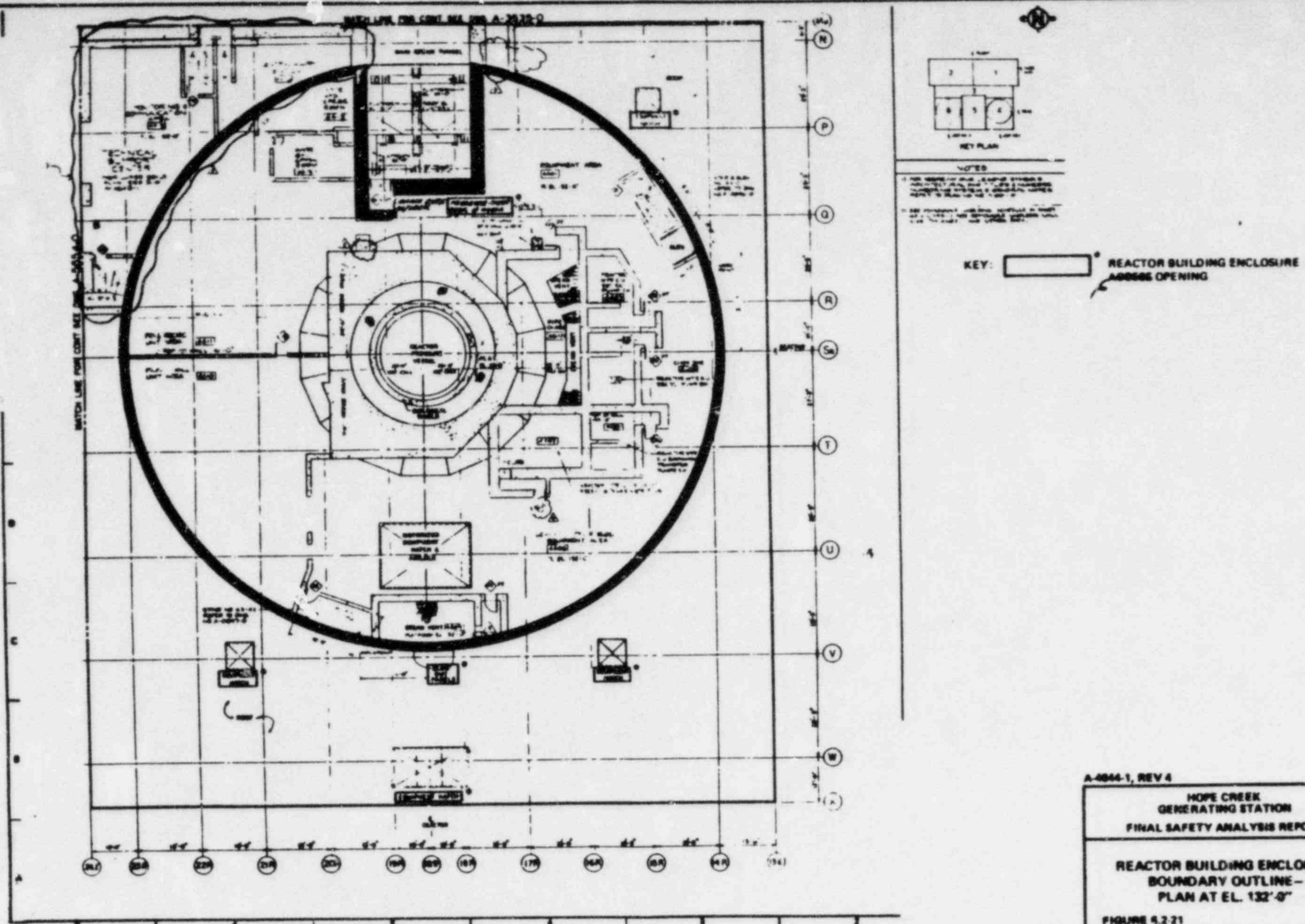

A-4844-1, REV 4

HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

REACTOR BUILDING ENCLOSURE
BOUNDARY OUTLINE–
PLAN AT EL. 132'-0"

FIGURE 6.2-21

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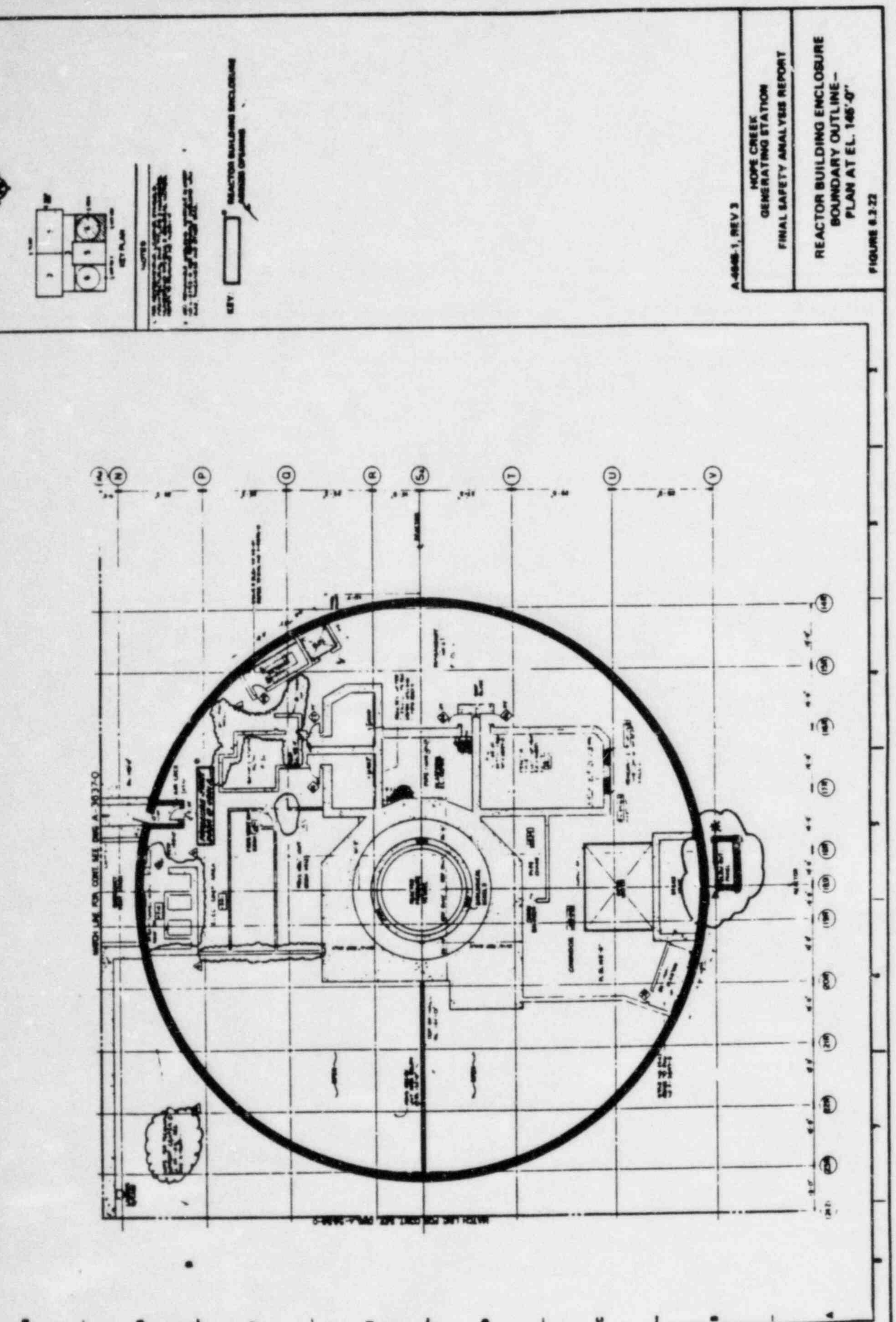

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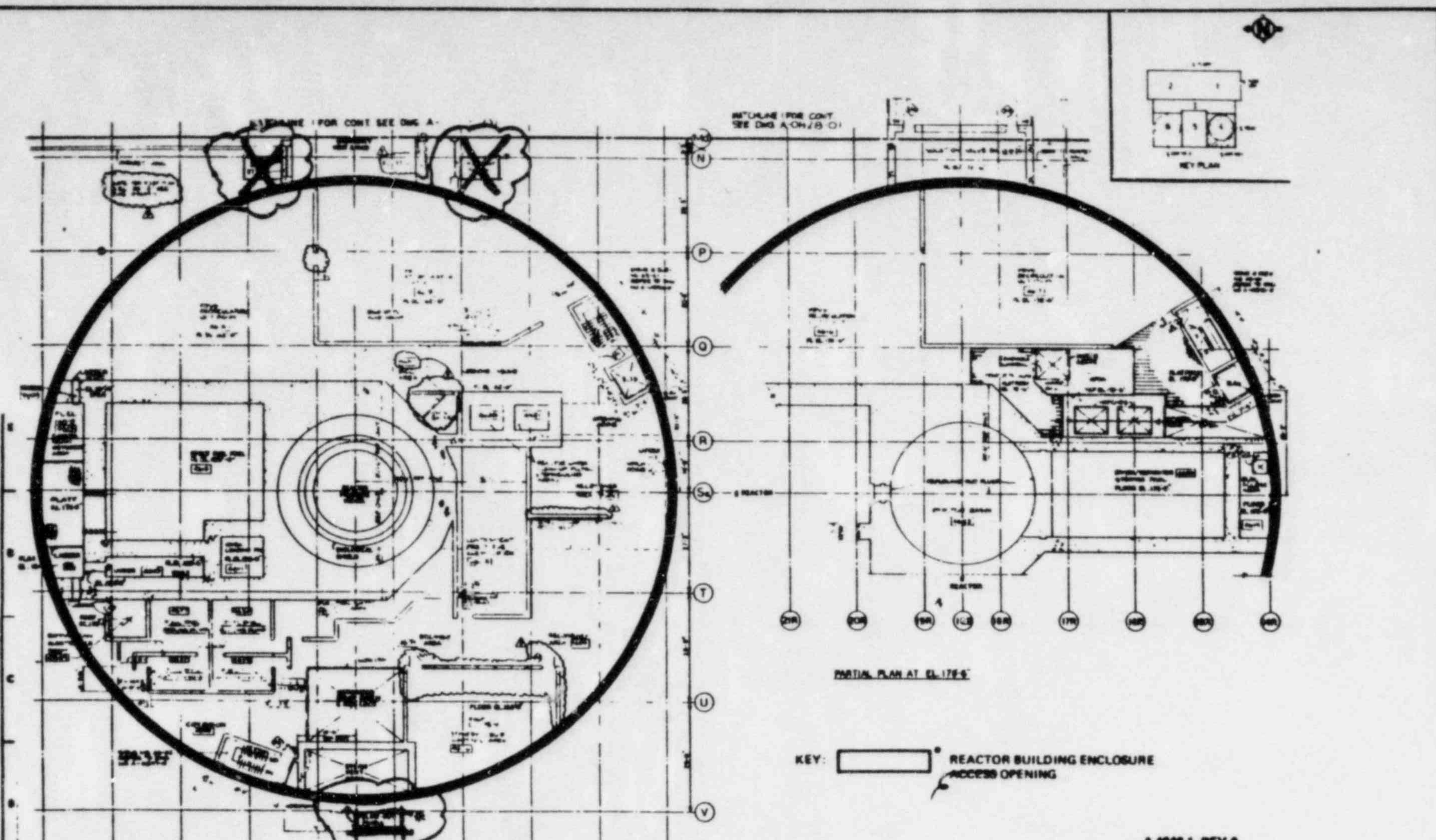
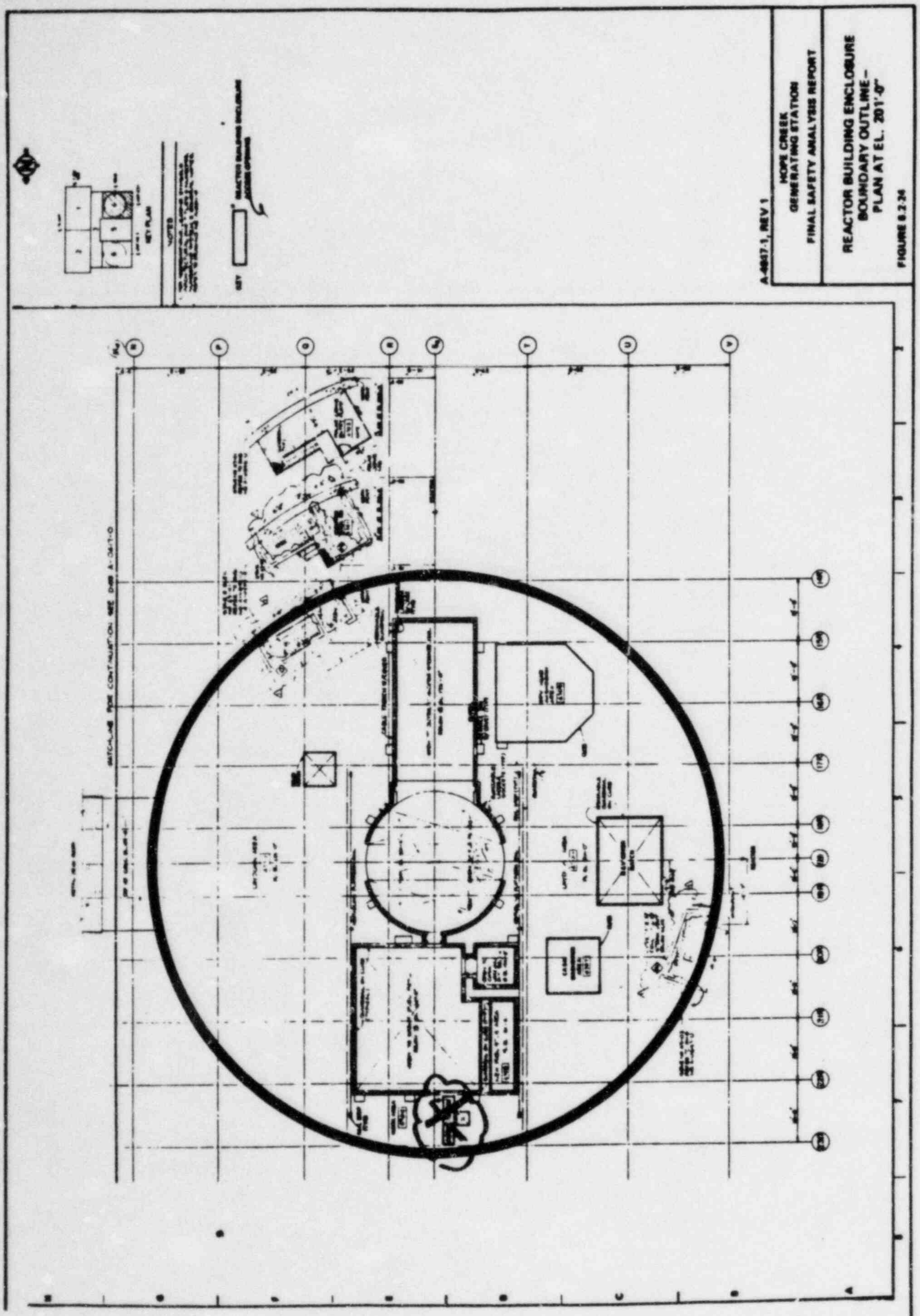


A-4046-1, REV. 0

HOPE CREEK GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

REACTOR BUILDING ENCLOSURE BOUNDARY OUTLINE– PLAN AT EL. 162'-0", 178'-6"

FIGURE 6.2-23



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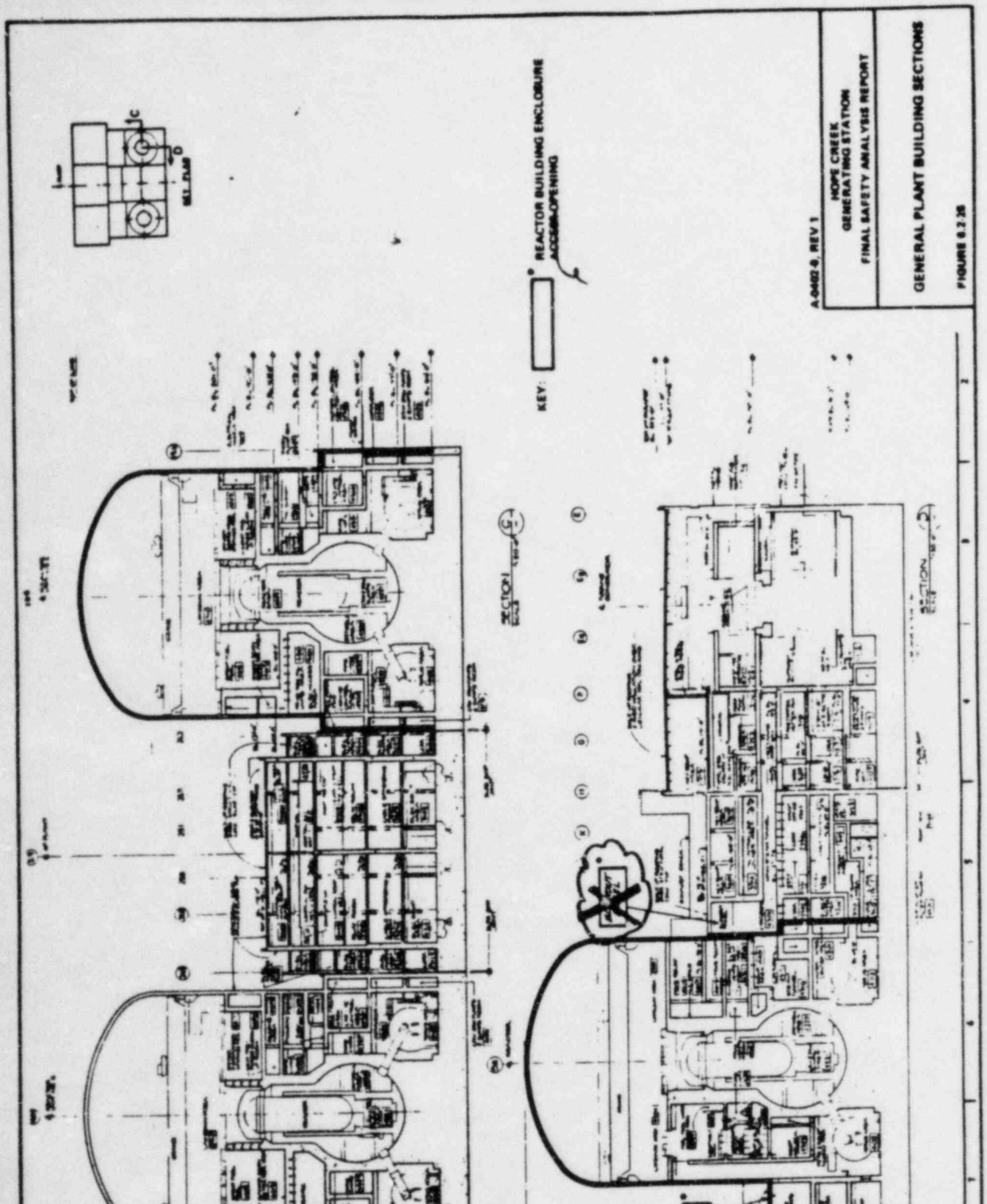

DSER OPEN ITEM 131

DSER OPEN ITEM 136 (Section 6.3.5, 15.9.13)

PLANT SPECIFIC LOCA ANALYSIS

The LOCA analyses reported in the FSAR were for a lead plant representative of Hope Creek. The applicant has committed to supply plant-specific LOCA analyses in a later amendment to the FSAR before fuel loading. The NRC staff will report the results of its review of the plant-specific analyses in a supplement to this report.

The applicant has included small-break LOCA calculations in FSAR Section 6.3.3 that were performed for a lead plant representative of Hope Creek. The applicant has committed to supply plant-specific LOCA analyses in a later amendment before fuel load. The staff will report on its review of the plant-specific analyses in a supplement to this report.

Response

The plant-specific LOCA analysis will be provided in July 1985 and will utilize the evaluation model described in Reference 1 and accepted by the NRC staff in Reference 2.

References

1. "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50, Appendix K," NEDE-20566P, November 1975.

2. Letter to G.G. Sherwood (General Elecric) from R.L. Tedesco (NRC), "Acceptance for Referencing of Topical Reports-20566P, NEDO-20566-1 Revision 1, and NEDE-20566-4 Amendment 4," February 4, 1981.

DSER Open Item No. 166 (Section 12.3)

AIRBORNE RADIOACTIVITY MONITOR POSITIONING

The applicant should clarify how he intends to use the ventilation monitors to accurately monitor plant iodine levels when the air being monitored by these monitors has been filtered through the plant HEPA and charcoal filter banks.

RESPONSE

FSAR Section 12.3.4.2.2 has been revised to address how HCGS intends to accurately monitor particulates and iodine from any compartment which has a possibility of containing airborne radioactivity and which normally may be occupied by personnel, taking into account dilution in the ventilation system.

taps are located in the ducts next to the detectors so that grab samples can be taken.

Additional mobile samplers with monitoring detectors that are displayed, controlled, and recorded by the CRP are provided for use if needed.

More details about airborne radioactive material sampling and monitoring are included in Section 11.5.

The above described airborne radioactive material monitoring equipment and procedures are used to meet the applicable parts of Regulatory Guides 1.21, 1.97, 8.2, 8.8, 8.12, and ANSI N13.1-1969.

Acceptance Criteria II.B.17 of standard review plan 12.3 - 12.4 provides criteria for the establishment of locations for fixed continuous area gamma radiation monitors. The specific document referenced is ANSI/ANS-HPSSC-6.8.1-1981. The locations and numbers of monitors used at HCGS are not in full compliance with this standard. The location of these monitors are in the vicinity of personnel access areas only. These locations are based on the dose assessment and operating experiences from other nuclear power plants. In addition, these locations were finalized prior to the issuance of this standard and provide an acceptable method of monitoring area radiation levels.

Insert →

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~~Acceptance Criterion II.4.b.3 requires ventilation monitors to be placed upstream of the HEPA filters. HCGS design places the ventilation monitors downstream of the HEPA filter in order to assess the plant's effluents. This is achieved best at this location as:~~

~~a. It is more efficient to have a single monitoring point rather than multiple points.~~

~~b. The instrument is sufficiently sensitive to ensure compliance with technical specification release limits.~~

~~c. The ventilation effluent monitors referred to above and the HVAC in line monitors (see P&IDs in Section 9.4) are scintillation detectors. These monitors are used to detect gross activity and as such will indicate~~

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delete

increases in airborn radioactivity concentrations. Maintenance of iodine concentration within 10 MPC-hours will be assured by the use of several methods including these monitors, in-plant surveys, and portable particulate and iodine sampling monitors. Grab samples may be obtained from the duct systems or the room air by using the portable samplers. These samples are then analyzed in the laboratory by multichannel analyzer (MCA). (See Section 12.5 for further information about MCA). Therefore, particulate and iodine sampling monitors are not provided upstream of the HEPA filters.

12.3.5 REFERENCES

12.3-1 J.J. Martin and P.H. Blichert-Toft, "Radioactive Atoms, Auger Electrons, and X-Ray Data," Nuclear Data Tables, Academic Press, October 1970.

12.3-2 J.J. Martin, Radioactive Atoms Supplement 1, ORNL 4923, Oak Ridge National Laboratory, August 1973.

12.3-3 W.W. Bowman and K.W. MacMurdo, "Radioactive Decays Ordered by Energy and Nuclide," Atomic Data and Nuclear Data Tables, Academic Press, February 1970.

12.3-4 M.E. Meek and R.S. Gilbert, Summary of X-Ray and Gamma- Ray Energy and Intensity Data, NEDO-12037, General Electric, January 1970.

13.3-5 C.M. Lederer, et al, Table of Isotopes, 6th edition, John Wiley, New York, 1967 (1st corrected printing March 1968).

12.3-6 D.S. Duncan and A.B. Spear, "Grace 1 - An IBM 704-709 Program Design for Computing Gamma Ray Attenuation and Heating in Reactor Shields," Atomics International, NAA-SR-3719, June 1959.

12.3-7 D.S. Duncan and A.B. Spear, "Grace 2 - An IBM 709 Program for Computing Gamma Ray Attenuation and

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Insert

Acceptance Criterion II.4.b.3 requires ventilation monitors to be placed upstream of HEPA filters. The HCGS design places scintillation detectors in ducts that are tributary to the release vent in order to provide warning of increased releases within the plant. These instruments detect increases in the gross noble gas concentrations of the effluent. Hence, placement of the detectors relative to HEPA and/or charcoal filters does not significantly affect their response. Since releases of iodines and particulates will be accompanied by much larger releases of noble gases, the changes in ventilation monitor readings provide indication of a change in airborne activity concentration in one or more of the plant's areas. If an increase is detected, its source and magnitude will be determined using portable samplers.

Normally occupied non-radiation areas in the plant do not have potential for significant airborne concentrations of particulates and iodine during plant operation because:

a. The ventilation systems are designed to prevent the spread of airborne radioactivity into normally occupied areas.

b. Highy radioactive piping/components are not located in normally occupied areas.

Certain activities, such as refueling, solid waste handling, or turbine teardown may increase the possibility of encountering significant airborne activities in some normally occupied areas. Continuous local airborne monitoring will be provided during these activities, as needed.

Exposure of personnel to high concentrations of airborne activity in radiation areas will be prevented through in-plant surveys and these portable particulate and iodine sampling monitors prior to personnel entrance. Continuous monitoring will be provided as required by area conditions and the nature of the entry. Administrative control will prevent inadvertent entry of personnel into normally unoccupied areas (Zone III and above). The provisions discussed above ensure that personnel will not be inadvertently exposed to significant concentrations of airborne activity.

Therefore, continuous ventilation radioactivity monitors capable of detecting 10 MPC-hrs of particulate and iodine from any normally occupied compartments are not provided as permanently installed equipment.

DSER Open Item No. 167 (Section 12.3.4.2)

PORTABLE CONTINUOUS AIR MONITORS

If portable continuous air monitors are used to monitor plant airborne radioactivity levels, the applicant should demonstrate that he has a sufficient number of CAMs to continuously monitor for both particulate and iodine radioactivity levels in all normally occupied locations where airborne radioactivity may exist (such as solid waste handling areas, the spent fuel pool area, the reactor operating floor, and the turbine building). This is an open item.

RESPONSE

Based on the response to DSER Open Item 166, portable CAMs will not normally be used to continuously monitor particulate and iodine activity in normally occupied areas.

DSER Open Item No. 169 (Section 12.5.3)

DIVISION 8 REG GUIDES

The applicant has not addressed how the guidance of the following Division 8 RGs, as given in SRP Section 12.5, will be followed. If the guidance of these guides will not be followed, the applicant should describe the specific alternative approaches to be used.

(1) RG 8.20 "Applications of Bioassay for I-125 and I-131."

(2) RG 8.26, "Applications of Bioassay for Fission and Activation Products."

(3) RG 8.27, "Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants."

(4) RG 8.28, "Audible Alarm Dosimeters."

(5) RG 8.29, "Instruction Concerning Risks From Occupational Radiation Exposure."

This is an open item.

RESPONSE

A change to the FSAR will be made per the attached mark-up to indicate HCGS commitment to the guidance contained in the referenced Regulatory Guides.

To ensure adequate manpower at all times for radiation protection supervisory functions, the experience and qualification requirements of the senior radiation protection supervisor and radiation protection supervisor positions may be reduced on a temporary basis. The general manager - Hope Creek operations approves or disapproves such action following review of the radiation protection engineer's recommendations.

The qualifications of the radiation protection technicians meet or exceed the personnel requirements of ANSI 3.1-1981. The technicians are supported by personnel in the radiation protection department in the assistant and worker classification.

12.5.2 FACILITIES, EQUIPMENT AND INSTRUMENTATION

Radiation protection facilities, equipment, and instrumentation were designed and acquired to meet the requirements of Regulatory Guides 1.97, 8.3, 8.4, 8.8, 8.9, 8.12, 8.14, ~~and~~ 8.15, and 8.28

12.5.2.1 Radiation Protection and Radiochemistry Facilities

12.5.2.1.1 Access Control

HCGS has two general area classifications for radiological control purposes: the restricted area and the radiologically controlled area (RCA). The restricted area is any area where access is controlled to protect all individuals from exposure to radiation or radioactive material. In general, the HCGS restricted area corresponds to the area inside the station security fence (protected area). The RCA, which is within the restricted area, features positive control over access, activities, and egress. Access is limited in accordance with operational requirements and individual training (in radiation protection). The RCA may include radiation areas, high radiation areas, contaminated areas, radioactive material storage areas, and airborne radioactivity areas. Entry to and exit from the permanent RCA is through two designated access control points. The access control points, shown on Figures 12.5-2 and 12.5-3, are located at elevations 124 and 137 feet in the service and radwaste areas of the auxiliary building. Self-survey personnel monitoring equipment, such as hand and foot, portal, or Geiger-Mueller (G-M) type friskers, are located at the exit from the RCA.

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used as a low volume grab air sample. The filter medium is removed and analyzed in more detail in the radiation protection counting room.

12.5.2.2.6 Personnel Protective Equipment

Special protective equipment such as coveralls, plastic suits, shoe covers, gloves, head covers, and respirators, including approved air purifying respirators, self-contained breathing apparatus (pressure demand) and airline respirators and hoods, are stored in various plant locations and clothing change areas. This equipment is used to prevent both deposition of radioactive material internally or on body surfaces and the spread of contamination. Most areas of the plant are kept free of contamination so that no special protective equipment is needed. Contaminated areas are identified with posted signs. Radiation signs and radiation/~~exposure~~ work permits (RWPs) are the primary means of defining the equipment required to enter work these contaminated areas.

A variety of combinations of protective equipment may be prescribed, depending on the nature and level of possible contamination. For example, cotton clothes may be adequate, but in wet areas, plastic rain suits or bubble suits may be prescribed. Respirators are required if airborne hazards exist, or if surface contamination could cause an airborne hazard as defined in the radiation protection procedures.

12.5.3 PROCEDURES

Radiation protection procedures, as described in this section, are implemented by Hope Creek radiation protection instructions, administrative procedures, ALARA procedures, and emergency plan procedures. The procedures are written to meet the guidelines of Regulatory Guides 1.8, 1.16, 1.33, 1.39, 8.2, 8.7, 8.8, 8.9, 8.10, ~~and~~ 8.13, 8.20, 8.26, 8.27 and 8.29

12.5.3.1 Radiological Surveys

Area survey procedures describe the purpose and techniques of detecting and measuring levels of radiation and contamination. Contamination may be on surfaces or airborne. Area surveys are conducted throughout the plant. Such surveys may be routine or may be related to specific jobs. An area survey may be performed

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DSER Open Item 177 (Section 15.1.1)

PARTIAL FEEDWATER HEATING

The applicant was asked to justify that operation with partial feedwater heating to extend the cycle beyond the normal end-of-cycle condition would not result in a more limiting change in minimum critical power ratio than that obtained using the assumption of normal feedwater heating. The staff requires that analyses be provided before operation in this mode if a decision is made to operate in this mode. Until such analyses are provided, the staff will condition the license from operation in this mode.

RESPONSE

PSE&G will accept a license condition as described above until such time as a HCGS analysis is submitted.