PSE&G Public Service Electric and Gas Company

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Robert L. Mittl General Manager
Nuclear Assurance and Regulation

September 7, 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 the Draft SER open items, and FSAR question responses listed in Attachment 3. Please note that the proposed change to FSAR Section 12.3.4 in response to DSER Item No. 166, covers radiation protection items discussed via a telecon between Charles Hinson (NRC-RAB) and Russell Lovell (PSE&G) regarding supplemental radiation monitoring. PSE&G is making a commitment to provide the additional information requested by July 1, 1985. It is our understanding that identifying the location of supplemental monitors may be handled as a comfirmatory item pending radiation protection/health physics inspections.

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Also, enclosed for your review (see Attachment 5) is a copy of the revised FSAR Section 1.10, Item II.K.3.25 as requested by G. Thomas of the Reactor Systems Branch.

In reviewing DSER Appendix A, PSE&G has identified the following discrepancies:

a. The tag number of the Reactor Water Cleanup Filter - Demineralizer Hoist is incorrectly entered in Table 2.1 of DSER Appendix A as 10H203. The correct tag number was 10H213. As described in the attached DSER responses, two new hoists, 1AH220 and 1BH220, have replaced 10H213.

b. The tag numbers of the Diesel Generator Underhung Crane, 1AH400 through 1DH400, are incorrectly entered in Table 2.1 as 0AH301 through 0DH301.

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

Very truly yours,

R J Mittl

Attachments/Enclosure

C D. H. Wagner
USNRC Licensing Project Manager

W. H. Bateman
USNRC Senior Resident Inspector

DATE: 9/7/84

ATTACHMENT 1

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
1	2.3.1	Design-basis temperatures for safety-related auxiliary systems	Complete	8/15/84
2a	2.3.3	Accuracies of meteorological measurements	Complete	8/15/84 (Rev. 1)
2b	2.3.3	Accuracies of meteorological measurements	Complete	8/15/84 (Rev. 1)
2c	2.3.3	Accuracies of meteorological measurements	Complete	8/15/84 (Rev. 2)
2d	2.3.3	Accuracies of meteorological measurements	Complete	8/15/84 (Rev. 2)
3a	2.3.3	Upgrading of onsite meteorological measurements program (III.A.2)	Complete	8/15/84 (Rev. 2)
3b	2.3.3	Upgrading of onsite meteorological measurements program (III.A.2)	Complete	8/15/84 (Rev. 2)
3c	2.3.3	Upgrading of onsite meteorological measurements program (III.A.2)	NRC Action	
4	2.4.2.2	Ponding levels	Complete	8/03/84
5a	2.4.5	Wave impact and runup on service water intake structure	Complete	9/7/84 (Rev. 2)
5b	2.4.5	Wave impact and runup on service water intake structure	Complete	9/7/84 (Rev. 2)
5c	2.4.5	Wave impact and runup on service water intake structure	Complete	7/27/84
5d	2.4.5	Wave impact and runup on service water intake structure	Complete	9/7/84 (Rev. 2)
6a	2.4.10	Stability of erosion protection structures	Complete	8/20/84
6b	2.4.10	Stability of erosion protection structures	Complete	8/20/84
6c	2.4.10	Stability of erosion protection structures	Complete	8/03/84

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ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
7a	2.4.11.2	Thermal aspects of ultimate heat sink	Complete	8/3/84
7b	2.4.11.2	Thermal aspects of ultimate heat sink	Complete	8/3/84
8	2.5.2.2	Choice of maximum earthquake for New England - Piedmont Tectonic Province	Complete	8/15/84
9	2.5.4	Soil damping values	Complete	6/1/34
10	2.5.4	Foundation level response spectra	Complete	6/1/84
11	2.5.4	Soil shear moduli variation	Complete	6/1/84
12	2.5.4	Combination of soil layer properties	Complete	6/1/84
13	2.5.4	Lab test shear moduli values	Complete	6/1/84
14	2.5.4	Liquefaction analysis of river bottom sands	Complete	6/1/84
15	2.5.4	Tabulations of shear moduli	Complete	6/1/84
16	2.5.4	Drying and wetting effect on Vincentown	Complete	6/1/84
17	2.5.4	Power block settlement monitoring	Complete	6/1/84
18	2.5.4	Maximum earth at rest pressure coefficient	Complete	6/1/84
19	2.5.4	Liquefaction analysis for service water piping	Complete	6/1/84
20	2.5.4	Explanation of observed power block settlement	Complete	6/1/84
21	2.5.4	Service water pipe settlement records	Complete	6/1/84
22	2.5.4	Cofferdam stability	Complete	6/1/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
23	2.5.4	Clarification of FSAR Tables 2.5.13 and 2.5.14	Complete	6/1/84
24	2.5.4	Soil depth models for intake structure	Complete	6/1/84
25	2.5.4	Intake structure soil modeling	Complete	8/10/84
26	2.5.4.4	Intake structure sliding stability	Complete	8/20/84
27	2.5.5	Slope stability	Complete	6/1/84
28a	3.4.1	Flood protection	Complete	8/30/84 (Rev. 1)
28b	3.4.1	Flood protection	Complete	8/30/84 (Rev. 1)
28c	3.4.1	Flood protection	Complete	8/30/84 (Rev. 1)
28d	3.4.1	Flood protection	Complete	8/30/84 (Rev. 1)
28e	3.4.1	Flood protection	Complete	8/30/84 (Rev. 1)
28f	3.4.1	Flood protection	Complete	7/27/84
28g	3.4.1	Flood protection	Complete	7/27/84
29	3.5.1.1	Internally generated missiles (outside containment)	Complete	8/3/84 (Rev. 1)
30	3.5.1.2	Internally generated missiles (inside containment)	Closed (5/30/84-Aux.Sys.Mtg.)	6/1/84
31	3.5.1.3	Turbine missiles	Complete	7/18/84
32	3.5.1.4	Missiles generated by natural phenomena	Complete	7/27/84
33	3.5.2	Structures, systems, and components to be protected from externally generated missiles	Complete	7/27/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
34	3.6.2	Unrestrained whipping pipe inside containment	Complete	7/18/84
35	3.6.2	ISI program for pipe welds in break exclusion zone	Complete	6/29/84
36	3.6.2	Postulated pipe ruptures	Complete	6/29/84
37	3.6.2	Feedwater isolation check valve operability	Complete	8/20/84
38	3.6.2	Design of pipe rupture restraints	Complete	8/20/84
39	3.7.2.3	SSI analysis results using finite element method and elastic half-space approach for containment structure	Complete	8/3/84
40	3.7.2.3	SSI analysis results using finite element method and elastic half-space approach for intake structure	Complete	8/3/84
41	3.8.2	Steel containment buckling analysis	Complete	6/1/84
42	3.8.2	Steel containment ultimate capacity analysis	Complete	8/20/84 (Rev. 1)
43	3.8.2	SRV/LOCA pool dynamic loads	Complete	6/1/84
44	3.8.3	ACI 349 deviations for internal structures	Complete	6/1/84
45	3.8.4	ACI 349 deviations for Category I structures	Complete	8/20/84 (Rev. 1)
46	3.8.5	ACI 349 deviations for foundations	Complete	8/20/84 (Rev. 1)
47	3.8.6	Base mat response spectra	Complete	8/10/84 (Rev. 1)
48	3.8.6	Rocking time histories	Complete	8/20/84 (Rev. 1)

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
49	3.8.6	Gross concrete section	Complete	8/20/84 (Rev. 1)
50	3.8.6	Vertical floor flexibility response spectra	Complete	8/20/84 (Rev. 1)
51	3.8.6	Comparison of Bechtel independent verification results with the design-basis results	Complete	8/20/84 (Rev. 2)
52	3.8.6	Ductility ratios due to pipe break	Complete	8/3/84
53	3.8.6	Design of seismic Category I tanks	Complete	8/20/84 (Rev. 1)
54	3.8.6	Combination of vertical responses	Complete	8/10/84 (Rev. 1)
55	3.8.6	Torsional stiffness calculation	Complete	6/1/84
56	3.8.6	Drywell stick model development	Complete	8/20/84 (Rev. 1)
57	3.8.6	Rotational time history inputs	Complete	6/1/84
58	3.8.6	"O" reference point for auxiliary building model	Complete	6/1/84
59	3.8.6	Overturning moment of reactor building foundation mat	Complete	8/20/84 (Rev. 1)
60	3.8.6	BSAP element size limitations	Complete	8/20/84 (Rev. 1)
61	3.8.6	Seismic modeling of drywell shield wall	Complete	6/1/84
62	3.8.6	Drywell shield wall boundary conditions	Complete	6/1/84
63	3.8.6	Reactor building dome boundary conditions	Complete	6/1/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
64	3.8.6	SSI analysis 12 Hz cutoff frequency	Complete	8/20/84 (Rev. 1)
65	3.8.6	Intake structure crane heavy load drop	Complete	6/1/84
66	3.8.6	Impedance analysis for the intake structure	Complete	8/10/84 (Rev. 1)
67	3.8.6	Critical loads calculation for reactor building dome	Complete	6/1/84
68	3.8.6	Reactor building foundation mat contact pressures	Complete	6/1/84
69	3.8.6	Factors of safety against sliding and overturning of drywell shield wall	Complete	6/1/84
70	3.8.6	Seismic shear force distribution in cylinder wall	Complete	6/1/84
71	3.8.6	Overturning of cylinder wall	Complete	6/1/84
72	3.8.6	Deep beam design of fuel pool walls	Complete	6/1/84
73	3.8.6	ASHSD dome model load inputs	Complete	6/1/84
74	3.8.6	Tornado depressurization	Complete	6/1/84
75	3.8.6	Auxiliary building abnormal pressure	Complete	6/1/84
76	3.8.6	Tangential shear stresses in drywell shield wall and the cylinder wall	Complete	6/1/84
77	3.8.6	Factor of safety against overturning of intake structure	Complete	8/20/84 (Rev. 1)
78	3.8.6	Dead load calculations	Complete	6/1/84
79	3.8.6	Post-modification seismic loads for the torus	Complete	8/20/84 (Rev. 1)

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
80	3.8.6	Torus fluid-structure interactions	Complete	6/1/84
81	3.8.6	Seismic displacement of torus	Complete	8/20/84 (Rev. 1)
82	3.8.6	Review of seismic Category I tank design	Complete	8/20/84 (Rev. 1)
83	3.8.6	Factors of safety for drywell buckling evaluation	Complete	6/1/84
84	3.8.6	Ultimate capacity of containment (materials)	Complete	8/20/84 (Rev. 1)
85	3.8.6	Load combination consistency	Complete	6/1/84
86	3.9.1	Computer code validation	Complete	8/20/84
87	3.9.1	Information on transients	Complete	8/20/84
88	3.9.1	Stress analysis and elastic-plastic analysis	Complete	6/29/84
89	3.9.2.1	Vibration levels for NSSS piping systems	Complete	6/29/84
90	3.9.2.1	Vibration monitoring program during testing	Complete	7/18/84
91	3.9.2.2	Piping supports and anchors	Complete	6/29/84
92	3.9.2.2	Triple flued-head containment penetrations	Complete	6/15/84
93	3.9.3.1	Load combinations and allowable stress limits	Complete	6/29/84
94	3.9.3.2	Design of SRVs and SRV discharge piping	Complete	6/29/84

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
95	3.9.3.2	Fatigue evaluation on SRV piping and LOCA downcomers	Complete	6/15/84
96	3.9.3.3	IE Information Notice 83-80	Complete	8/20/84 (Rev. 1)
97	3.9.3.3	Buckling criteria used for component supports	Complete	6/29/84
98	3.9.3.3	Design of bolts	Complete	6/15/84
99a	3.9.5	Stress categories and limits for core support structures	Complete	6/15/84
99b	3.9.5	Stress categories and limits for core support structures	Complete	6/15/84
100a	3.9.6	10CFR50.55a paragraph (g)	Complete	6/29/84
100b	3.9.6	10CFR50.55a paragraph (g)	Complete	8/20/84
101	3.9.6	PSI and ISI programs for pumps and valves	Complete	8/20/84
102	3.9.6	Leak testing of pressure isolation valves	Complete	6/29/84
103a1	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103a2	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103a3	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84
103a4	3.10	Seismic and dynamic qualification of mechanical and electrical equipment	Complete	8/20/84

ATTACHMENT 1 (Cont'd)

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

ATTACHMENT 1 (Cont'd)

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

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
112c	5.2.5	Reactor coolant pressure boundary leakage detection	Complete	8/30/84 (Rev. 1)
112d	5.2.5	Reactor coolant pressure boundary leakage detection	Complete	8/30/84 (Rev. 1)
112e	5.2.5	Reactor coolant pressure boundary leakage detection	Complete	8/30/84 (Rev. 1)
113	5.3.4	GE procedure applicability	Complete	7/18/84
114	5.3.4	Compliance with NB 2360 of the Summer 1972 Addenda to the 1971 ASME Code	Complete	7/18/84
115	5.3.4	Drop weight and Charpy v-notch tests for closure flange materials	Complete	9/5/84 (Rev. 1)
116	5.3.4	Charpy v-notch test data for base materials as used in shell course No. 1	Complete	7/18/84
117	5.3.4	Compliance with NB 2332 of Winter 1972 Addenda of the ASME Code	Complete	8/20/84
118	5.3.4	Lead factors and neutron fluence for surveillance capsules	Complete	8/20/84
119	6.2	TMI item II.E.4.1	Complete	6/29/84
120a	6.2	TMI Item II.E.4.2	Complete	8/20/84
120b	6.2	TMI Item II.E.4.2	Complete	8/20/84
121	6.2.1.3.3	Use of NUREG-0588	Complete	7/27/84
122	6.2.1.3.3	Temperature profile	Complete	7/27/84
123	6.2.1.4	Butterfly valve operation (post accident)	Complete	6/29/84

ATTACHMENT 1 (Cont'd)

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

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
134	6.2.6	Containment leakage testing	Complete	6/15/84
135	6.3.3	LPCS and LPCI injection valve interlocks	Complete	8/20/84
136	6.3.5	Plant-specific LOCA (see Section 15.9.13)	Complete	8/20/84 (Rev. 1)
137a	6.4	Control room habitability	Complete	8/20/84
137b	6.4	Control room habitability	Complete	8/20/84
137c	6.4	Control room habitability	Complete	8/20/84
138	6.6	Preservice inspection program for Class 2 and 3 components	Complete	6/29/84
139	6.7	MSIV leakage control system	Complete	6/29/84
140a	9.1.2	Spent fuel pool storage	Complete	9/7/84 (Rev. 2)
140b	9.1.2	Spent fuel pool storage	Complete	9/7/84 (Rev. 2)
140c	9.1.2	Spent fuel pool storage	Complete	9/7/84 (Rev. 2)
140d	9.1.2	Spent fuel pool storage	Complete	9/7/84 (Rev. 2)
141a	9.1.3	Spent fuel cooling and cleanup system	Complete	8/30/84 (Rev. 1)
141b	9.1.3	Spent fuel cooling and cleanup system	Complete	8/30/84 (Rev. 1)
141c	9.1.3	Spent fuel pool cooling and cleanup system	Complete	8/30/84 (Rev. 1)

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
141d	9.1.3	Spent fuel pool cooling and cleanup system	Complete	8/30/84 (Rev. 1)
141e	9.1.3	Spent fuel pool cooling and cleanup system	Complete	8/30/84 (Rev. 1)
141f	9.1.3	Spent fuel pool cooling and cleanup system	Complete	8/30/84 (Rev. 1)
141g	9.1.3	Spent fuel pool cooling and cleanup system	Complete	8/30/84 (Rev. 1)
142a	9.1.4	Light load handling system (related to refueling)	Complete	8/15/84 (Rev. 1)
142b	9.1.4	Light load handling system (related to refueling)	Complete	8/15/84 (Rev. 1)
143a	9.1.5	Overhead heavy load handling	Complete	9/7/84
143b	9.1.5	Overhead heavy load handling	Open	
144a	9.2.1	Station service water system	Complete	8/15/84 (Rev. 1)
144b	9.2.1	Station service water system	Complete	8/15/84 (Rev. 1)
144c	9.2.1	Station service water system	Complete	8/15/84 (Rev. 1)
145	9.2.2	ISI program and functional testing of safety and turbine auxiliaries cooling systems	Closed (5/30/84-Aux.Sys.Mtg.)	6/15/84
146	9.2.6	Switches and wiring associated with HPCI/RCIC torus suction	Closed (5/30/84-Aux.Sys.Mtg.)	6/15/84

ATTACHMENT 1 (Cont'd)

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

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
157	9.5.1.4.e	Cable tray protection	Complete	8/20/84
158	9.5.1.5.a	Class B fire detection system	Complete	6/15/84
159	9.5.1.5.a	Primary and secondary power supplies for fire detection system	Complete	6/1/84
160	9.5.1.5.b	Fire water pump capacity	Complete	8/13/84
161	9.5.1.5.b	Fire water valve supervision	Complete	6/1/84
162	9.5.1.5.c	Deluge valves	Complete	6/1/84
163	9.5.1.5.c	Manual hose station pipe sizing	Complete	6/1/84
164	9.5.1.6.e	Remote shutdown panel ventilation	Complete	6/1/84
165	9.5.1.6.g	Emergency diesel generator day tank protection	Complete	6/1/84
166	12.3.4.2	Airborne radioactivity monitor positioning	Complete	9/7/84 (Rev. 1)
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

M P84 80/12 16 - gs

ATTACHMENT 1 (Cont'd)

OPEN ITEM	DSER SECTION NUMBER	SUBJECT	STATUS	R. L. MITTL TO A. SCHWENCER LETTER DATED
174	13.5.2	Resolution explanation in FSAR of TMI Items I.C.7 and I.C.8	Complete	6/15/84
175	13.6	Physical security	Open	
176a	14.2	Initial plant test program	Complete	8/13/84
176b	14.2	Initial plant test program	Complete	9/5/84 (Rev. 1)
176c	14.2	Initial plant test program	Complete	7/27/84
176d	14.2	Initial plant test program	Complete	8/24/84 (Rev. 2)
176e	14.2	Initial plant test program	Complete	7/27/84
176f	14.2	Initial plant test program	Complete	8/13/84
176g	14.2	Initial plant test program	Complete	8/20/84
176h	14.2	Initial plant test program	Complete	8/13/84
176i	14.2	Initial plant test program	Complete	7/27/84
177	15.1.1	Partial feedwater heating	Complete	8/20/84 (Rev. 1)
178	15.6.5	LOCA resulting from spectrum of postulated piping breaks within RCP	NRC Action	
179	15.7.4	Radiological consequences of fuel handling accidents	NRC Action	
180	15.7.5	Spent fuel cask drop accidents	NRC Action	
181	15.9.5	TMI-2 Item II.K.3.3	Complete	6/29/84
182	15.9.10	TMI-2 Item II.K.3.18	Complete	6/1/84
183	18	Hope Creek DCRDR	Complete	8/15/84

ATTACHMENT 1 (Cont'd)

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

ATTACHMENT 1 (Cont'd)

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

ATTACHMENT 1 (Cont'd)

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

<u>ATTACHMENT 1 (Cont'd)</u>

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

ATTACHMENT 1 (Cont'd)

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

ATTACHMENT 1 (Cont'd)

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

ATTACHMENT 1 (Cont'd)

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

ATTACHMENT 1 (Cont'd)

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

M P84 80/12 25- gs

ATTACHMENT 2 DATE: 9/7/84

DRAFT SER SECTIONS AND DATES PROVIDED

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

Notes:

1. Open items provided in letter dated July 24, 1984 (Schwencer to Mittl)

2. Open items provided in June 6, 1984 meeting

3. Open items provided in April 17-18, 1984 meeting

4. Open items provided in May 2, 1984 meting

5. Draft SER Section provided in letter dated August 7, 1984 (Schwencer to Mittl)

CT:db

MP 84 95/03 01

DATE: 9/7/84

ATTACHMENT 3

Open Item	DSER Section	Subject
5a	2.4.5	Wave impact and runup on service water intake structure
5b	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
140a	9.1.2	Spent fuel pool storage
140b	9.1.2	Spent fuel pool storage
140c	9.1.2	Spent fuel pool storage
140d	9.1.2	Spent fuel pool storage
143a	9.1.5	Overhead heavy load handling
166	12.3.4.2	Airborne radioactivity monitor positioning
Question 421.10		

ATTACHMENT 4

DSER Open Item No. 5 (DSER Section 2.4.5)

WAVE IMPACT AND RUNUP ON SERVICE WATER INTAKE STRUCTURE

The applicant has analyzed the wind waves that would traverse plant grade coincident with the PMH surge hydrograph and runup on safety-related facilities. These calculations were based on the assumption that wind waves would be generated in the Delaware Estuary and progress to the site. As the surge level would begin to rise, resulting from the approaching eye of the postulated hurricane, the wind speed would progressively change direction from the southeast clockwise to the west. Waves encroaching on the southern end of the Island would be depth-limited (i.e., the waves would "feel" bottom and thus become shallow water waves) by plant grade elevation on both the Salem and Hope Creek sites. These depth-limited (shallow water) waves will impact and runup on the southern and western faces of the safety-related structures in the power block. The applicant has stated that the southern face of the Reactor Building and the Auxiliary Building are designed for a flood protection level of 38.0 ft msl or 3.2 ft above the maximum calculated wave runup height of 34.8 ft msl and the other exposures of safety-related structures have a flood protection level of 32.0 ft msl or 1 ft above the maximum calculated wave runup height of 31.0 ft msl.

The staff has requested the applicant to provide additional information on the waves that impact on the river face of service water intake structure. The waves impacting on this face of the structure are not reduced in height (depth-limited) as those that traverse plant grade.

As indicated in Section 2.4.1, the applicant states that all accesses to safety-related structures (doors and hatches) are provided with water-tight seals designed to withstand the head of water associated with the flood protection levels. But, the applicant has not indicated whether the water-tight doors are designed to withstand either the combined loading effects of both static water level and the dynamic wave impact or, as cited in Sections 3.4.1 and 3.5.1.4 of this report, the impact of a barge propelled by winds and waves associated with a hydrologic event that floods plant grade.

Based upon its analysis according to SRP 2.4.5, the staff concludes that the flood protection level of El. 38.0 ft msl for the southern face of the Reactor Buidling and Auxiliary Building and El. 32.0 ft msl for the remaining safety-related structures within the power block meets the requirements of Regulatory Guide 1.59. Until additional information and analysis

DSER Open Item No. 5 (Cont'd)

are available, the staff cannot conclude that the flood protection level of El. 32.0 ft msl for the Service Water Intake Structure meets the requirements of Regulatory Guide 1.59. Based on its analysis, the staff cannot conclude that the plant meets the requirements of GDC 2 with respect to the hydrologic aspects of Probable Maximum Surges and Seiche Flooding.

RESPONSE

The requested information for the service water intake structure has been provided in the responses to the following NRC questions:

Information Provided	Question No.
Wave runup elevations	240.8
Wave impact loads	240.9
Flood protection	240.8 and 410.69

As a result of discussions with the NRC staff, the response to Question 410.69 has been revised and summary calculations for wave overtopping of the west and south walls have been submitted under separate cover.

Information on the ability of the doors and hatches to withstand the combined loading effects of static water level and the dynamic wave impact is provided in the response to FSAR Question 240.14.

QUESTION 410.69 (Section 9.2.1)

Provide a figure(s) in the FSAR which shows the protection of the station service water system from the flood water (including wave effects) of the design basis flood.

RESPONSE

The general arrangement of the intake structure is provided in Figures 1.2-40 and 1.2-41. Section AA of Figure 1.2-41 is reproduced here as Figure 410.69-1 which identifies the watertight areas and the walls and slabs designed to accommodate flood loads. As described in Sections 2.4.2 and 2.4.5, the south and west exterior walls of the intake structure are subject to a maximum wave run-up elevation of 134.4 feet due to the probable maximum hurricane (PMH). Such waves could overtop the roof of the western portion of the structure at elevation 128 feet. However, a rigorous analysis has been performed to determine the depth of water in the low area (elevation 122.0 feet) after wave impact and to confirm that water does not enter the building through the air intake control dampers (bottom elevation 128.5 feet). Therefore, flood water will not enter into the dry area of the intake structure. On the north side of the intake structure, the maximum water level will be only slightly higher than the still water elevation (113.8 feet) during the PMH. According to Table 2.4.6, the maximum wave elevation for the north side of the intake structure is 26.3 feet MSL (elevation 115.3 feet) due to a postulated multiple dam break. Therefore, flood protection of the north exterior wall to elevation 121.0 feet is adequate.

On the east side of the intake structure, the maximum wave run-up elevation due to the PMH equals 122.3 feet. This elevation is due to a 1% wave traveling in the direction of Fetch "A". Fetch A, which is rotated about 15 degrees from Fetch 1 (as shown in Figures 410.69-2 and 410.69-3), is chosen to maximize the wave run-up elevation. Elevation 122.3 feet exceeds the elevation of the bottom of the HVAC exhaust openings at elevation 122.0 feet by 0.3 feet. Curbs will be added at the bottom of these openings to prevent water from entering into the building.

In addition the following assessments have been made to confirm the adequacy of the structure and interior components for the overtopping wave:

a. The exterior walls are designed to withstand the flood loads including the dynamic wave action effects.

b. The roof hatches at both elevations 122.0 and 128.0 feet have been sealed (caulking, gaskets, etc.) to prevent any intrusion of water. The hatch covers are keyed into

RESPONSE - cont'd

the openings to prevent any adverse slippage due to wave induced loadings.

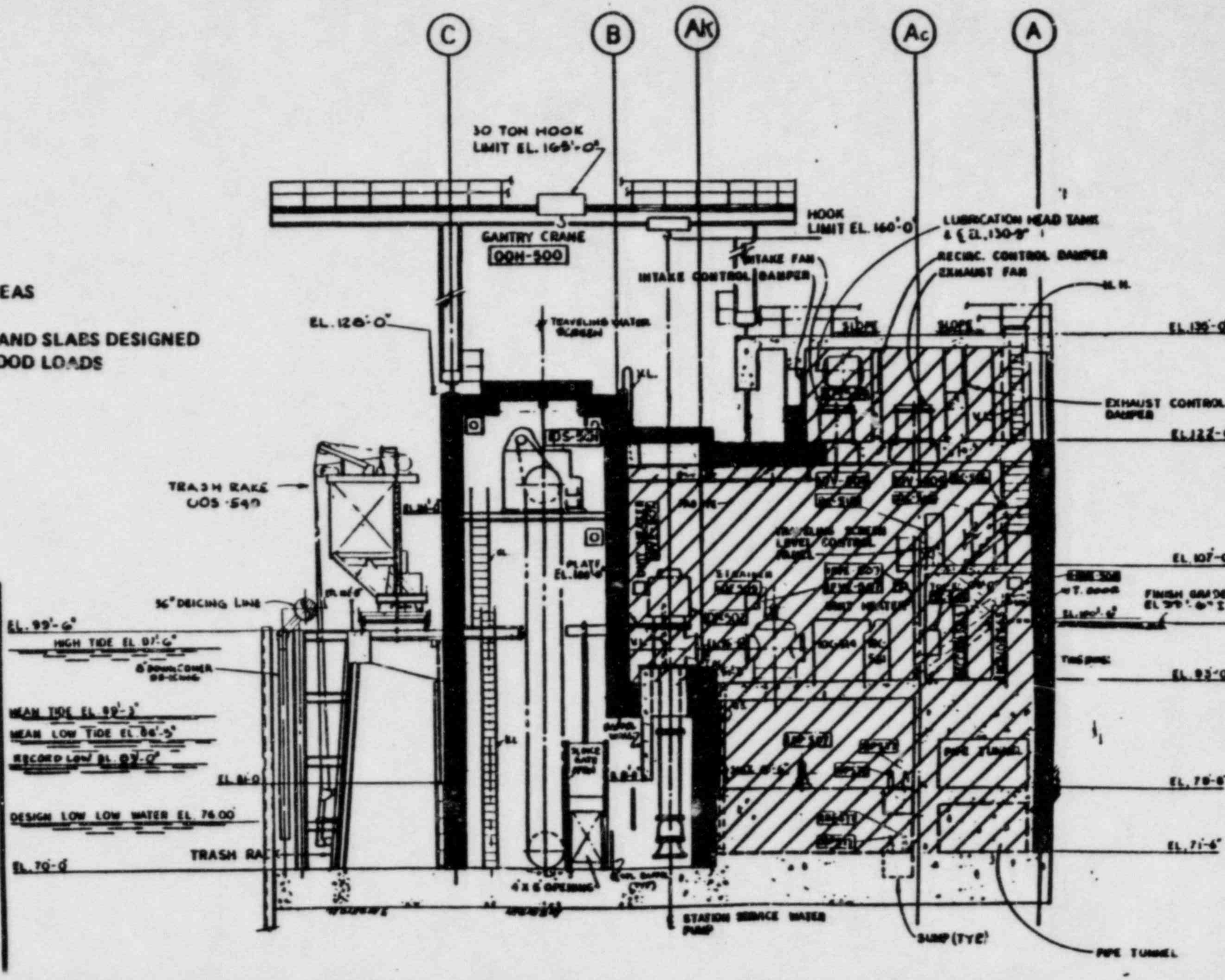
c. All Seismic Category I components except for the traveling water screens are located within the dry areas of the structure.

d. The traveling water-screens, located in the "wet" area between column lines B and C have electric motors which are fully protected against the flood water level.

e. A condition was postulated where suspended moisture enters the dry areas of the structure through the air intake control dampers. It has been assessed that all of the Seismic Category I components subjected to this environment will continue to function as required.

Section 3.4.1 and Table 3.4-1 have been revised for clarification.

SEP -5 '84



LEGEND:

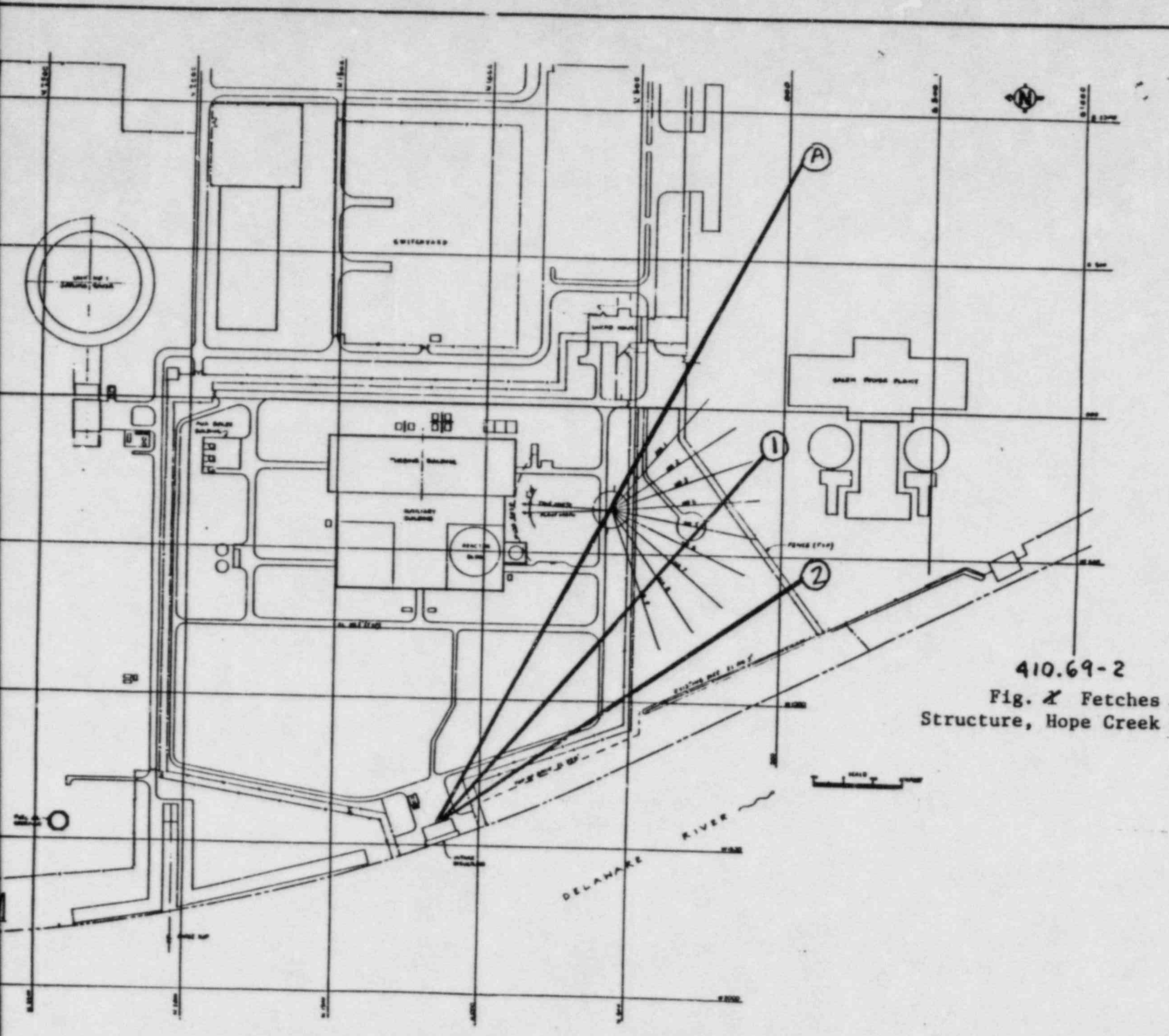
DRY AREAS

WALLS AND SLABS DESIGNED FOR FLOOD LOADS

HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

SERVICE WATER INTAKE STRUCTURE — FLOOD PROTECTION

FIGURE 410.69-1



410.69-2

Fig. X Fetches for Service Water Intake Structure, Hope Creek Generating Station.

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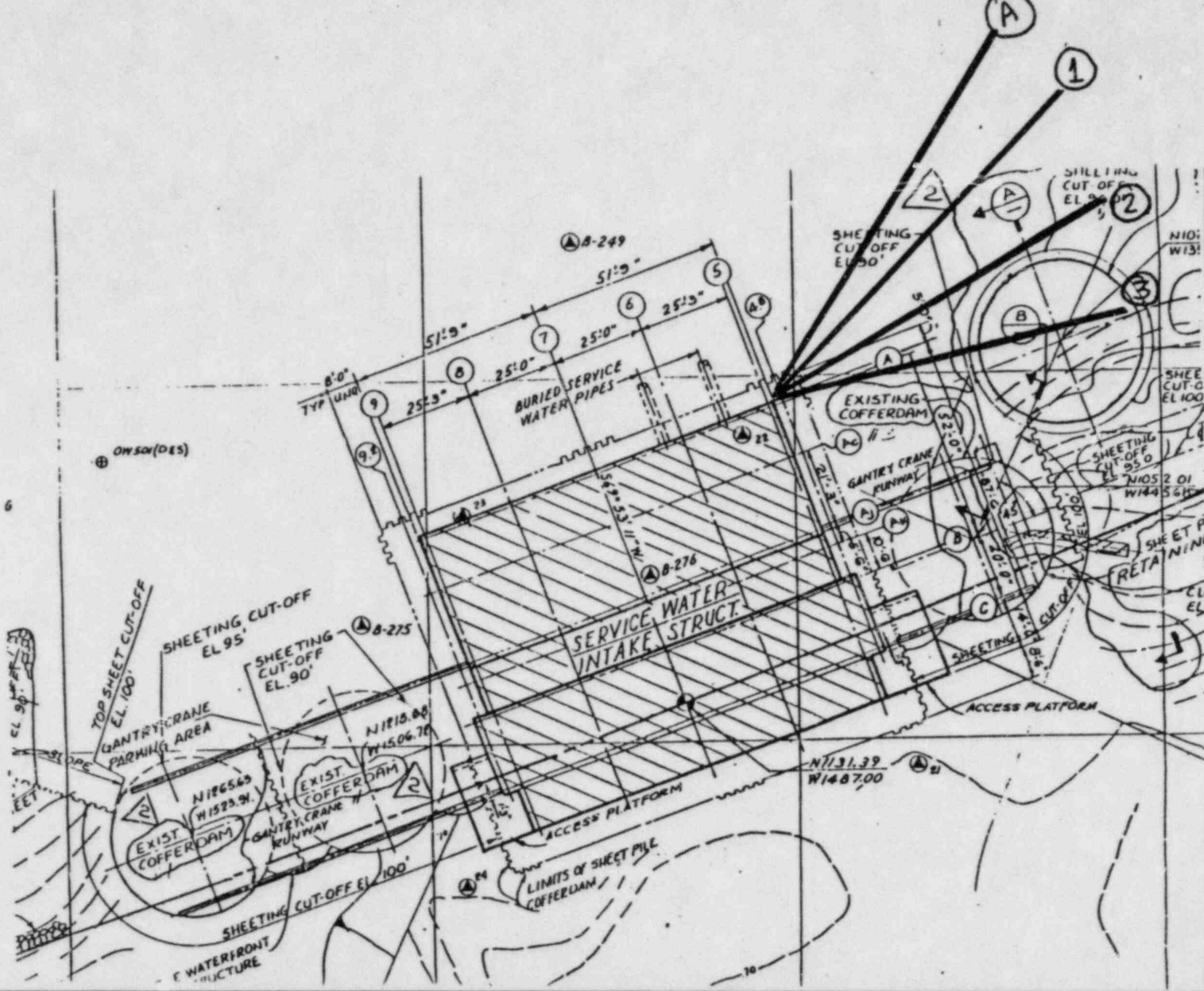
410.69-3

Fig. 2. Fetches for Service Water Intake Structure, Hope Creek Generating Station



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SEP -5 '84 0270628

August 6, 1984

Hope Creek Generating Station
Analysis of Overtopping of Service Water Intake Structure

I. Wave Calculations

- o Wave heights and periods as well as still-water levels and runup elevations are as given in Table 2.4-10a of FSAR (Amendment 5, April 1984).

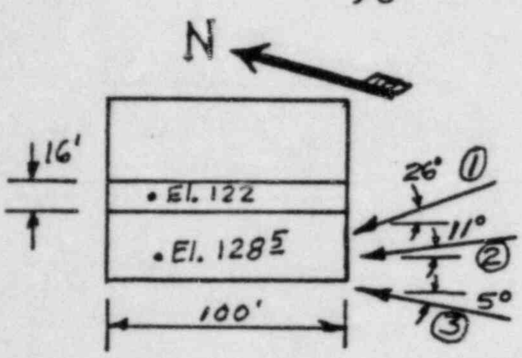
II. Overtopping Calculations

- o Overtopping rates were calculated for west face and south face where top of wall elevations are 128.5 and 122.0, respectively.

- o Equations from Wiegel (1976) were used for the overtopping calculations.

$$Q = (g\,Q_o^* H_o'^3)\exp\left(-\frac{0.217}{2\alpha}\log_e\left(\frac{R+h-d_s}{R-h+d_s}\right)\right)$$

$$Q_o^* = \frac{\left(\frac{\epsilon}{2\pi}\right)^2\left(\frac{H}{H_o'}\right)^2 \tanh\left(\frac{2\pi d_s}{L}\right)}{\frac{H_o'}{gT^2}}$$



- o where ϵ was taken as $1/2\pi$ in order to maximize the value of Q_o^* (see Figure 6 of Wiegel's paper)

- o α was taken as 0.06 in order to maximize Q (see Equation 4 of Wiegel's paper).

- o Conservative assumptions in calculating overtopping rates were:

  - It was assumed that waves attacked normal to the wall of the structure.

  - It was assumed that the train of waves was made up of all 1% waves.

  - It was assumed that wave height was constant along the crest.

- o Calculated overtopping rate was increased to allow for wind speed using Equation (7-11) of the 1977 edition of the U. S. Army Corps of Engineers Shore Protection Manual.

$$K' = 1.0 + W_f\left(\frac{h-d_s}{R} + 0.1\right)\sin\theta$$

- In making the wind adjustment the factor W_f was assumed to be 2.0 for onshore winds greater than 60 mph. The angle O was 90°.

- o After adjustment for wind the overtopping rates were adjusted for angle of attack by multiplying the overtopping rate by the sin of the angle between the fetch vector and the wall.

III. Maximum water surface elevations were calculated by backwater calculation starting from the north end of the roof.

- o The separate overtopping rates were added and the total was assumed to flow off the top of the structure at the north end.

- o Critical depth was assumed to occur at the downstream end of the channel and was calculated as:

$$y_c = \left[\frac{(Q_{TOT}/16)^2}{32.2}\right]^{1/3}$$

where Q is the rate of flow from the west side in cfs/ft.

- o The backwater calculation assumes a gradually varied steady flow.

$$y_{x+\Delta x} = \sqrt{\frac{2\Delta Q \cdot \Delta x \cdot Q_x}{16 \cdot 32.2 \cdot y_x} + y_x^2}$$

- o Calculations were performed moving upstream starting with the depth at the north end.

- o The calculations showed that fetch 3 was the critical case. The total flow rate for fetch 3 was 0.5 cfs/ft from the west and 14.7 cfs/ft from the south end.

- o The maximum water surface elevation reached was 126.9 for the fetch 3 condition which is well below the critical 128.5 elevation at which flow could enter the air intakes.

IV. A separate calculation was made considering a surge generated by flow coming over the south end of the building. The depth of flow and velocity of flow ahead of the surge resulting from the previous surge had to be assumed. Velocity ahead of the surge was assumed to be zero, since that condition maximizes the surge height. Depth ahead of the surge was assumed to be 1.0' and does not have a really significant affect on the height of the following surge. The resulting elevation of the crest of the generated surge was 126.9 which is below the 128.5 elevation at which water can flow into the air intake.

V. A check was made to see if flow could surge into the air intakes as a result of plunging from the roof at elevation 128.5.

- o Loss coefficients of 0.5 at the entrance to the air intake opening and 0.5 at the bend (see attached sketch were assumed).

- o Velocity at the edge of the 128.5 elevation roof section was calculated assuming critical depth there and was increased by 50% for reasons of conservancy.

- o The velocity approaching the entrance to the air intake chamber was calculated using the energy equation and neglecting losses.

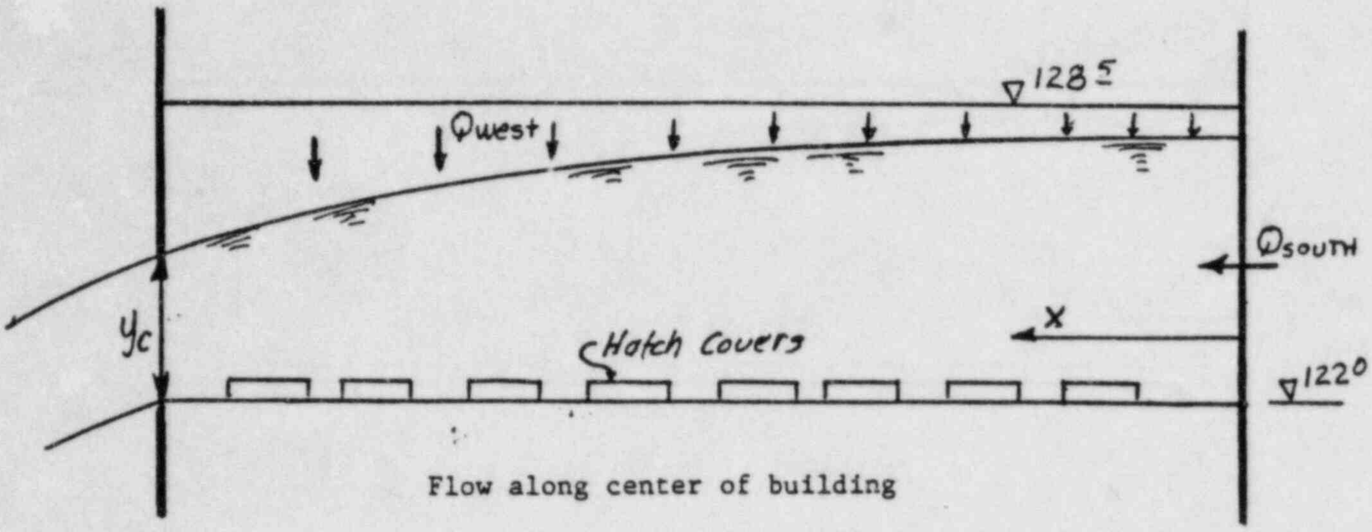
- o Losses incurred by turbulence and impact of the jet entering water ponded on top of elevation 122.0 were neglected.

- o Headloss through the screens was neglected.

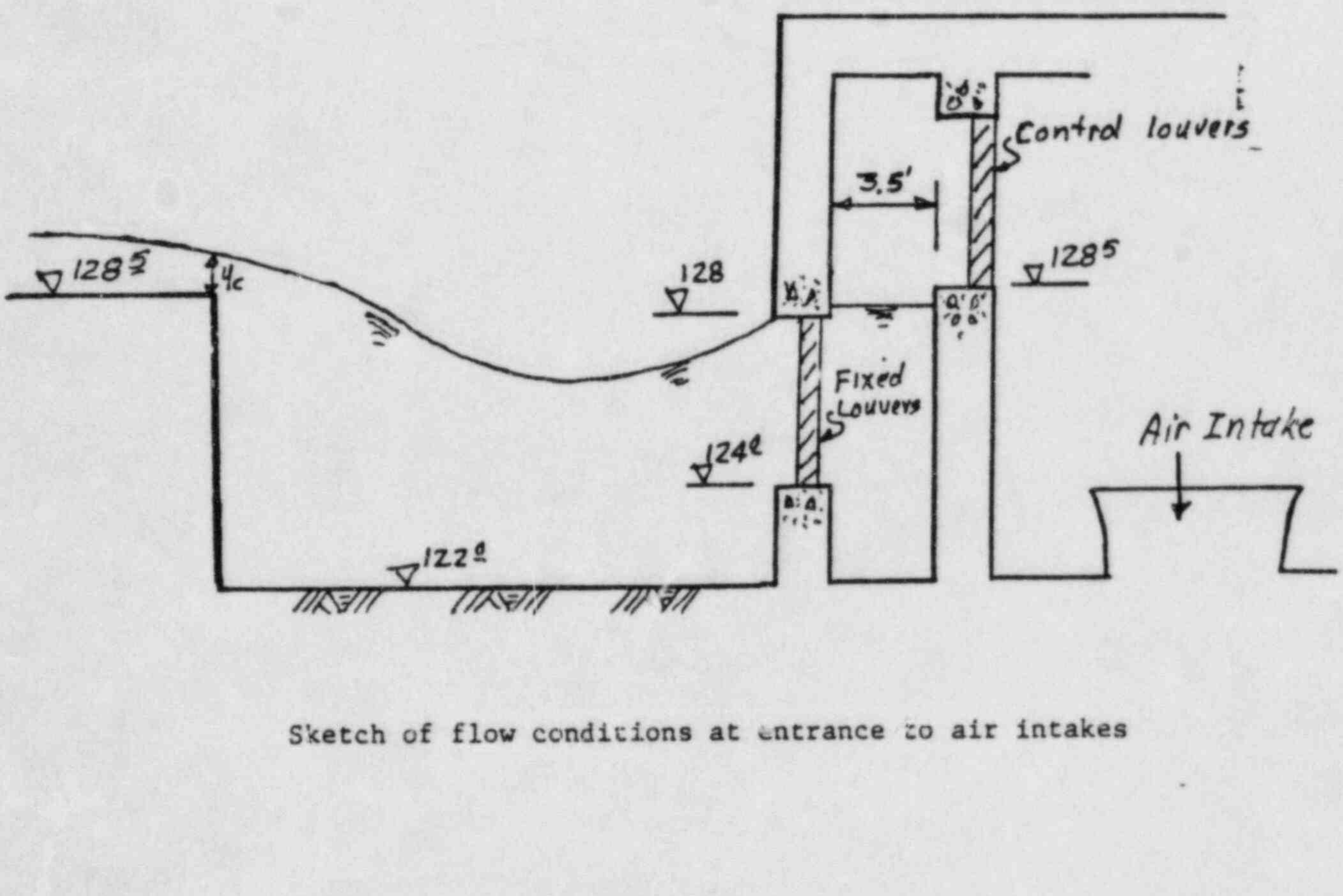
- o The maximum elevation achieved was calculated to be 126.3 or well below the 128.5 elevation at which water could flow into the building.

- o A separate analysis was made using a one-dimensional momentum approach. The presence of the louver on top of the outer wall was neglected. A velocity of 26 feet per second was assumed to occur over the top of the lower outer wall whose top elevation is at 124.0. This velocity was calculated assuming that the total potential energy in a wave runup to 134.4 would be converted to kinetic energy at elevation 124 without energy loss. The one-dimensional energy analysis, assuming a flow rate of 5.75 cfs/foot indicates that the water surface within the intake could rise to elevation 127.0 which is below the 128.5 elevation at which water could flow into the service-water intake structure. The assumption of a flow rate of 5.75 cfs/foot is very conservative since that is the total overtopping rate from the west side of the structure for the critical fetch conditions assuming the wave strikes normal to the structure wall.

- o The total pressure of the air intake fans equals 4.5 inches of water. The maximum elevations of 126.3 feet and 127.0 feet given above result in margins of 2.2 and 1.5 feet respectively with respect to the 128.5 feet elevation at which water could flow into the building. Therefore, there is sufficient margin to accommodate a rise in water level due to fan suction pressure.



Flow along center of building



Sketch of flow conditions at entrance to air intakes

References

1. Wiegel, R. L., "Wave Overtopping Equation" Proceedings of the 1976 Coastal Engineering Conference.

2. Jackowski, R. A. (Editor) Shore Protection Manual, U. S. Army Corps of Engineers, Coastal Engineering Research Center, 1977.

DSER Open Item No. 140 (DSER Section 9.1.2)

SPENT FUEL STORAGE

Since the applicant's application for an operating license was docketed in 1983, which is after the November 17, 1977 date specified in the SRP, the applicant must provide the results of an analysis which shows that a failure of the liner plate as a result of an SSE will not cause any of the following: (1) significant releases of radioactivity due to mechanical damage to the fuel; (2) significant loss-of-water from the pool which could uncover the fuel and lead to release of radioactivity due to heat up; (3) loss of the ability to cool the fuel due to flow blockage caused by a portion of one or more complete section of the liner plate falling on the top of the fuel racks; (4) damage to safety-related equipment as a result of the pool leakage; and (5) uncontrolled release of significant quantities on radioactive fluids to the environs; in accordance to the Standard Review Plan. These buildings are also designed against flooding and tornado missiles (refer to Section 3.4.1 and 3.5.2 of this SER). We cannot conclude that the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guides 1.13, "Spent Fuel Storage Facility Design Basis," Position C.3, 1.29, "Seismic Design Classification," Positions C.1 and C.2, have been met.

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

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

DSER Open Item No. 140 (Cont'd)

We cannot conclude that the spent fuel storage facility is in conformance with the requirements of General Design Criteria 2, 61, and 62 as they relate to protection of the spent fuel against natural phenomena, radiation protection, and prevention of criticality and the guidelines of Regulatory Guides 1.13, Positions C.1, C.3, and C.4 and 1.29, Positions C.1 and C.2, relating to the facility's design basis and seismic classification. The spent fuel storage facility does not meet the acceptance criteria of SRP Section 9.1.2. We will report resolution of this item in a supplement to this SER.

Additionally, the information provided through Amendment 3 was not sufficient for the staff to complete the evaluation of the compatibility and chemical stability of materials wetted by spent fuel pool water. To complete the review, the following information is requested:

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

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

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

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

RESPONSE

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

RESPONSE (Cont'd)

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

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

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

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

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

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

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

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

RESPONSE (Cont'd)

Similar rack designs, with vented Boral poison in stainless steel racks, have been licensed and have proven successful. HCGS's maximum anticipated radiation exposure for the Boral is 1.0×10^{11} rads. This radiation exposure assumes freshly discharged fuel assemblies are stored in each cell for a 20 year period and then replaced with freshly discharged fuel for a second 20 year period. Brooks and Perkins Product Performance Report No. 624 documents Boral's capability to withstand exposure of 1.0×10^{11} rads gamma and 5.3×10^{19} neutrons per sq. cm. in demineralized water without detectable outgassing attributable to Boral, decrease in neutron attenuation, nor any discernable physical changes. This testing was performed at the Phoenix Memorial Laboratory of the University of Michigan using the Ford Nuclear Reactor. Ongoing tests have exposed Boral to accumulated radiation doses up to 7×10^{11} rads. These specimens were also found to be structurally sound and neutron attenuation capabilities were not degraded by irradiation.

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

PSE&G will develop a program to monitor the Boral surveillance program of either Fermi, Monticello or Brown's Ferry by March, 1985. The response to Question 281.14 has been revised to reflect this response.

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

3.8.4.8.3 Spent Fuel Rack Design

Acceptance Criterion II.4.f requires that the spent fuel racks be designed in compliance with Appendix D of SRP 3.8.4, which requires that construction materials should conform to Section III, Subsection NF of the ASME Code.

INSERT C

The spent fuel racks are constructed of ASTM A-240 and ASTM A-564 stainless steel. The A-240 and A-564 material specifications are identical to the ASME SA-240 and SA-564 material specifications. All rack steel is supplied with certified material test reports.

The rack materials are procured under a Q.A. Program that is intended to comply with:

a. 10CFR50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants".

b. ANSI/ASME N45.2, "Quality Assurance Program Requirements for Nuclear Facilities", and

c. ANSI/ASME NQA-1, "Quality Assurance Program Requirements for Nuclear Power Plants".

3.8.5 FOUNDATIONS

Foundations for all Seismic Category I structures and the turbine building and the administration facility, which are non-Seismic Category I structures, are described in this section.

3.8.5.1 Description of the Foundations

The configuration of the foundation mats for the various structures is shown on Figure 3.8-37.

Reinforced concrete mat foundations are provided for all structures. Except for the station service water system (SSWS) intake structure, the mats rest either on the Vincentown Formation or on engineered structural backfill placed on the Vincentown Formation. The mat and the lean concrete leveling

Insert C

The design, analysis and fabrication of the spent fuel racks conforms with the applicable provisions of Subsection NF. See Appendix 9B for a description of the design, analysis and construction of the racks.

DSER OPEN ITEM 140

CHAPTER 9

TABLES

DSER OPEN ITEM 140

CHAPTER 9

FIGURES

1. **Normal storage conditions exist when the fuel storage racks are located in the pool and are covered with about 25 feet of water for radiation shielding, and with the maximum number of fuel assemblies or bundles in their design storage position.**

2. An abnormal storage condition may result from bundle accidental dropping of an ~~empty~~ fuel ~~rack,~~ or from damage caused by the horizontal movement of fuel handling equipment without first disengaging the fuel from the hoisting equipment.

b. It is assumed that the storage array is infinite in all directions. Since no credit is taken for leakage, the values reported as effective neutron multiplication factors are in reality infinite neutron multiplication factors. The biases between the calculated results and experimental results and the uncertainty involved in the calculations, as well as other uncertainties, are taken into account as part of the calculational procedure to ensure that the specified K_{eff} limits are met.

c. The racks are designed to protect the fuel assemblies from physical damage caused by impact from fuel assemblies. The rack design would prevent the release of radioactive materials in excess of 10 CFR 20 and 10 CFR 100 allowances under normal and abnormal storage conditions.

d. The racks are constructed in accordance with the QA requirements of 10 CFR 50, Appendix B.

e. The spent fuel storage racks are constructed in accordance with Seismic Category I requirements. The applicable code for the design of racks is ASME Section III, Subsection NF.

f. Spent fuel storage space is provided in the fuel storage pool to accommodate 5.3 core loads of fuel assemblies.

9.1.2.2.2.2 High Density Spent Fuel Storage Racks

High density spent fuel storage racks in the fuel pool store spent fuel transferred from the reactor vessel. These are top-entry racks.

The spent fuel storage racks are of freestanding design and are not attached to either the fuel pool wall or the fuel pool liner plate. The racks are constructed of stainless steel, and the details of rack construction will be provided prior to fuel load.

See Appendix 9B for a description of the design, analysis and construction of the ~~high density~~ spent fuel storage racks.

i. The maximum stress in the fully loaded rack in a faulted condition will be provided prior to fuel load.

j. The spent fuel storage racks also have the capability of storing control rod guide tubes, control rods, and defective fuel containers. When the spent fuel is stored in the spaces provided for storing the above the K_{eff} does not exceed 0.95.

k. Several design features reduce the possibility of heavy objects dropping into the fuel pool. The main and auxiliary hoists of the reactor building polar crane are single-failure proof. In addition, the main hoist is physically prevented from traveling in the truncated segment shown on Figure 9.1-31 by mechanical stops on the girders of the polar crane. The crane design is discussed in Section 9.1.5. The removable guardrail and the four-inch curb around the refueling cavities further limit the possibility of heavy objects dropping into the fuel pool.

l. The fuel storage pool has water shielding for the stored spent fuel. Liquid level sensors are installed to detect a low pool water level. Makeup water is available to ensure that the fuel will not be uncovered should a leak occur.

m. Since the fuel racks are made of noncombustible material and are stored underwater, there is no potential fire hazard. The large water volume also protects the spent fuel storage racks from potential pipe breaks and associated jet impingement loads.

9.1.2.3.3 INSERT A

9.1.2.4 Spent Fuel Rack Inservice Inspection

~~An inservice inspection program is in effect throughout the life of the racks to ensure that the quality of the poisoned racks is~~

Insert B

~~maintained. Test coupons are used for an ongoing inspection program.~~

9.1.2.4.1 Test Coupon Description and Installation

Details of test coupon description and installation will be provided prior to fuel load.

9.1.2.5 SRP Rule Review

In SRP Section 9.1.2, Acceptance Criterion II.1 requires conformance to ANS 57.2, Paragraph 5.1.1, which states that the spent fuel storage facility, including its equipment and safety-related structures, shall be designed to Seismic Category I requirements.

The spent fuel liner plates are non-Seismic Category I and are not considered safety-related. These liner plates are welded to Seismic Category I embeds in the pool walls. Their primary functions are to minimize pool leakage and facilitate decontamination of the pool walls. Since they are essentially nonload-bearing, they will not adversely affect the structural integrity of the fuel pool and the spent fuel storage racks, and therefore do not have to comply with Seismic Category I requirements. Any pool wall attachments will always be affixed to the wall embeds.

Acceptance Criterion II.6, ANS 57.2, Paragraph 5.4.1 states that at least one radiation monitor with audible alarm should be installed on the fuel handling machine.

At HCGS, permanent radiation monitors scanning the entire refueling floor are mounted on the reactor building walls. These monitors indicate and actuate audible alarms locally and in the control room. In addition, portable health physics instrumentation will be installed on the fuel handling platform whenever the refueling machine is used over the spent fuel pool and the reactor core. The radiation monitoring system, including the portable platform mounted health physics instrumentation, is considered to be adequate for protection of personnel in the reactor building during all phases of station operation.

9.1.2.3.3 CRITICALITY ANALYSIS AND RESULTS

The criticality analysis was performed using the input parameters contained in Table 9.1-19. Figure 9.1-20 shows the reference geometry used in the criticality analysis, the zero flux boundary and the postulated dropped fuel assembly.

The criticality analysis is based on new fuel with a nominal, flat U-235 enrichment of 3.4 w/o. No credit is taken for the burnable poison fuel rods which may be present in the fuel assemblies. The analysis simulates an infinite array of fuel of infinite length. The analysis uses Utility Associates International's (UAI's) diffusion theory model, CHEETAH-B/CORC-BLADE/PDQ7 as the main working model. The analysis includes the various criticality safety-related aspects of the rack design, including various sensitivity calculations. The Monte Carlo transport model, AMPX/KENO -IV, is used as the verification model to verify the reactivity of the nominal rack design.

UAI performed similar criticality analyses for Limerick and Susquehanna. # The analysis includes all the normal, abnormal, and accident conditions described in Section 9.1.2.3.1.

Table 9.1-20 summarizes the nominal value of K effective of the racks under normal, abnormal, and accident conditions. The various uncertainties and biases considered in the analysis are also included.

CALCULATIONAL MODELS

This section presents a description of the calculational models and the basic assumptions used in this criticality analysis.

The Working Model

The criticality analysis for the Hope Creek BWR spent fuel racks employs the CHEETAH-B/CORC-BLADE/PDQ-7 model as the basic engineering tool. CHEETAH-B[1] is UAI's BWR lattice code based on the original LEOPARD code and uses a modified ENDF/B-II cross section library. CORC-BLADE[2] generates equivalent diffusion theory cross sections for the control blade. The PDQ-7[4] program is the well-known few-group spatial diffusion theory code widely used by the industry. The CHEETAH-B/CORC-BLADE/PDQ-7 model, which is also a part of the LEAHS (Lifetime Evaluation and Analysis of Heterogeneous Systems) nuclear analysis series of Control Data Corporation, has been extensively tested through benchmarking calculations of measured criticals as well as through core physics calculations for several operating power reactors.

A zero current boundary condition was applied to the four sides of the unit reference storage rack cavity to produce an infinite array effect. The two-dimensional, PDQ-7 calculations were made for four neutron energy groups, two mesh intervals per fuel pin, a flat U-235 enrichment description and a zero axial buckling to simulate infinite fuel length.

The Verification Model

The verification calculation employs the KENO-IV[5]/AMPX[6] model. The basic neutron cross section data comes from the master libraries of AMPX - a 123 group GAM-THERMOS neutron library prepared from ENDF/B version II data. The NITAWL module of the AMPX program is used to perform a Nordheim integral treatment of the U-238 resonances accounting for the self-shielding effect. The working library produced by the NITAWL/AMPX module retains the 123 group energy structure and is used directly by KENO-IV.

In the KENO-IV calculation, the spent fuel rack geometry including each fuel and water rod cell is represented discretely. To simulate the arrangement of a large number of storage rack units, and for a non-leakage condition in the axial directions, a specular reflective condition is applied to all six sides of the reference case storage rack cavity ~~(Figure 2)~~.

~~3.4~~ Basic Assumptions

To ensure that the analysis follows a conservative approach and conforms to the general guidelines of criticality safety analysis in Reference ~~4~~ 9.1-10, the calculations are performed with the following assumptions:

1. A flat 3.4 w/o distribution in an 8x8 bundle, with U-234 neglected
2. Fresh fuel, no burnable poison
3. Minor structural members replaced by water, i.e., spacer grids
4. Fresh water
5. Fuel is channeled.

9.1.2.3.3.1 ~~4.0~~ REFERENCE CASE CALCULATIONS

Physical Parameters and the Basic Storage Rack Cavity Geometry

The reference storage rack cavity ~~(Figure 2)~~ has a pitch of 6.308" ± 0.030". The stainless steel canister has a nominal inside clearance of 6.080 to accommodate 8x8 fuel assembly channeled in 0.080" thick Zircaloy-4. Plates of the neutron absorber material Boral, consisting of B_4C in an aluminum matrix core and clad with an aluminum sheath, are fastened to the outside of the canister. The Boral plate has a nominal total thickness of 95 mils and a minimum B-10 density of 0.028 g/cm^2. Table ~~1~~ 9.1-19 contains the values of the input parameters used in the analysis.

The rack must accommodate both channeled and unchanneled fuel. Studies reveal that the channeled fuel in the rack is more reactive than the unchanneled fuel. Taking the conservative approach, the study here involves channeled fuel (except in the accident condition where the dropped fuel is unchanneled in order to permit the closest contact between the dropped fuel assembly and the rack).

Two small, but non-conservative changes were made to the reference case in order to facilitate modeling. First, the boral width was set at 4.48" instead of 4.465". Second, the stainless steel flanges used in welding the outer wrapper to the inner can were deleted. An adjustment was made using PDQ to account for these differences.

~~4.2~~ Results of the Reference Case Calculations

Using the input data from Table ~~1~~ 9.1-19 and Figure ~~2~~ 9.1-30 (except as noted above), the K_{eff} values of the reference case at 68°F were calculated for the calculational model described in Section ~~3.0.~~ The results are:

DSER OPEN ITEM 140

HCGS FSAR

	PDQ-7	KENO-IV
k_{eff}, reference calculation	0.9229	0.9306 ± 0.0042
95% confidence interval	______	0.9222 - 0.9390

Insert A page 6 of 16

HCGS FSAR

9.1.2.3.3.2

[redacted] SENSITIVITY AND TOLERANCE REACTIVITY CALCULATIONS

[redacted] Temperature Effect

Using the reference storage rack cavity geometry, the temperature of the fuel and pool water was varied. In addition to the nominal 68°F, 40°F and 212°F were studied and the results of the CHEETAH-B/CORC-BLADE/PDQ-7 runs are given on Table 9.1-20 [redacted] As shown, reactivity decreases continuously as temperature increases from 40°F.

[redacted] Void Effect

The effect of boiling (assuming equal voids inside and outside of the rack) was studied by varying the voids from 0% to 20% at a temperature of 212°F with the reference geometry. The CHEETAH-B/CORC-BLADE/PDQ-7 results are shown in [redacted] Table 9.1-20 [redacted]. As indicated, k_{eff} decreases continuously as the void fraction increases.

[redacted] Pitch Sensitivity

The rack design permits the storage cavity pitch to differ from the 6.308" nominal value by ±0.030". The pitch sensitivity calculations of this analysis show the reactivity effect of these tolerance components as well as the reactivity pitch sensitivity by expanding the calculational range from -0.060" to +.030" at .030" intervals. The results, which are [redacted] tabulated in Table 9.1-20 [redacted], indicate that in the neighborhood of the nominal pitch, the pitch reactivity coefficient is about .15%Δk per .030" pitch change.

[redacted]

DSER OPEN ITEM 140

Insert A page 7 of 10

HCGS FSAR

[redacted] Effect of Boron

The Boral Plates which separate two adjacent fuel assemblies have a nominal thickness of .095" (consisting of an 73 mil core and 11 mil aluminum sheaths) a nominal width of 4.465" and an overall length of 11 feet 3 inches. The minimum B-10 loading is 0.028 g/cm^2.

(a) Boron Width Tolerance

The effect of reducing the Boral width was examined. The PDQ-7 calculation for the reference case configuration with the Boral width reduced by 0.0625" yielded $k_\infty = 0.92641$. Hence, the reactivity increases due to the -0.0625" tolerance on Boral width is $\Delta k = +0.0029$.

(b) Boron Density

The boron density was maintained at .028 g/cm^2 for all calculations. This areal density is the minimum density allowed by manufacturing design specifications.

(c) Boral Core Thickness Variation

The sensitivity to the Boral core thickness was determined by calculations in which the thickness varied from 61 mils to 80 mils (the aluminum sheaths were varied within tolerance to obtain the worst case core thickness). The results, tabulated in Table 9.1-30 [redacted] show a continuous increase in reactivity as the core thickness increases. This is due to the fact that the areal density is held constant, so an increase in thickness reduces volumetric density and, to a small degree, the boral effectivenes:

DSER OPEN ITEM 140

[redacted] Dimensional and Positional Tolerances

The total Δk bias for dimensional and positional tolerances are calculated from five separate contributions:

(i) Pitch Reduction
(ii) Boral Width Reduction
(iii) Inter-Cavity Spacing Reduction
(iv) Off-center Loading
(v) Boral Thickness Increase

(i) Pitch Reduction. The effect of reducing the center-to-center spacing of the rack cavities is obtained from [redacted] Table 9.1-20 and is [redacted] $\Delta k_1 = 0.0015$.

(ii) Boral Width Reduction. The Δk bias due to reducing the Boral width by its tolerance, 0.0625" is obtained from [redacted] Table 9.1-20 and is $\Delta k_2 = 0.0029$.

(iii) Inter-Cavity Spacing Reduction. Any seismic effect that may reduce the separation distance between adjacent cavities can be determined from the pitch sensitivity study [redacted]. Bringing two adjacent cavities closer by 0.048" results in the canisters touching and a reactivity increase $\Delta k_1 = 0.0023$ (from Table 9.1-20 [redacted]). Since this reduction is the maximum reduction of pitch possible in this design, this effect will not be added to item (i), but will replace it.

(iv) Off-Center Loading. The free space existing between a properly center fuel assembly and the top casting allows an assembly to be loaded off-center in a cavity. It was shown that this condition causes no adverse reactivity effect since the resulting k_{eff} for off-centered loading is less than that for properly centered assemblies.

(v) <u>Boral Thickness Increase</u>. The worst case boral core thickness reactivity effect calculated due to manufacturing tolerance stackup (.080") is obtained from Table 9.1-20 [redacted] and is $\Delta k_5 = .0001$

The above positive Δk contributions are statistically combined to give the total Δk bias for mechanical and seismic uncertainties.

$$\Delta k = \sqrt{(\Delta k_1)^2 + (\Delta k_2)^2 + (\Delta k_5)^2} = 0.0037$$

9.1.2.3.3.3
[redacted] SPECIAL CASES

[redacted] Grappler Drop Accident

The accident considered is the inadvertent drop of the assembly grappler used in lifting assemblies within the spent fuel pool. In this accident, the grappler is dropped in such a way that assemblies in adjacent rack cavities are displaced such that they are resting in an off-center loading arrangement. The reactivity effect for this off-center arrangement was discussed in Section [redacted] ([redacted]
9.1.2.3.3.2(iv)

[redacted] Assembly Drop Accident

(a) Single Assembly Dropped on Top of Rack. No adverse reactivity effect is expected from dropping a fuel assembly on top of a fully loaded storage rack during fuel handling because of the large water thickness (~14 inches) existing between the top of the assemblies already inside the cavities and the dropped assembly resting on top of the rack. Moreover the PDQ-7 model assumes an infinite fuel length in the axial direction.

(b) Single Assembly Next to Rack. The dropping of an assembly outside the rack is a possible event because of the unobstructed water area existing between the periphery of the storage racks and the side walls of the pool.

A conservative analysis to evaluate this situation is illustrated in Figure [redacted]. An assembly, presumed to be
9.1-20

[redacted]

Insert A page 11 of 16

HEGS FSAR

dropped during handling, lodges paralled to an assembly in the outer cavity with no Boral slab separating the two assemblies. The dropped assembly is unchanneled to permit the closest contact with the rack[redacted] The dimensions used are those of the reference case. This arrangement of the dropped fuel assembly with a 3 1/2 x 3 finite fuel rack is reflected on three sides as indicated in Figure 9.1-20; the fourth side is a zero flux boundary. The k_{eff} result for this case was 0.9128. The result for the same geometry without the dropped fuel was 0.9064 giving an increase of reactivity of $\Delta k = 0.0063$ for the above dropped assembly configuration. [redacted]

6 Assembly Moving Between Two Storage Racks

The rack structural design does not allow sufficient room to fit a fuel assembly between any two of the high density spent fuel racks. Therefore, the movement of assemblies between racks is precluded.

DSER OPEN ITEM 140

9.1.2.3.3.4 New Fuel Storage in the Spent Fuel Racks

The feasibility of storage of fresh fuel in the high density spent fuel racks was analyzed. Storage of new fuel in the mist, partly flooded, and dry conditions are addressed below.

Insert A page 15 of 16
HCGS FSAR

25% Mist Condition

The storage of new fuel of uniform 3.4 w/o U-235 enrichment in the high density spent fuel rack in a 25% aqueous mist environment was analyzed with the KENO model ~~(refer to Figure 2)~~ 9. The resulting k_∞ and 95% confidence interval are shown below:

	K_∞	95% Confidence Interval
25% Mist	.6375±.0054	.6267- .6483

Dry Condition

UAI experience in the analysis of poisoned rack criticality indicates that the fully flooded rack configuration is the most reactive with reacitivy decreasing with a decrease in moderator density. The 25% mist condition analysis confirms this as shown below. For this reason a dry condition analysis was not performed since it too will be less reactive than the flooded condition.

Moderator Density	K_∞	95% Confidence Interval
Reference Case: 1.00 g/cm^3	.9306±.0042	.9223-.9389
25% Mist Condition:0.25 g/cm^3	.6375±.0054	.6267-.6483

Partly Flooded Condition

The totally flooded condition as analyzed in the reference case is more reactive than that of the partly flooded condition.

9.1.2.3.3.5 Special Spent Fuel Rack Storage

A 5x6 non-horated special rack is to be installed in the ~~Hope Creek~~ HCGS spent fuel pool. Storage of control rods, control rod guide tubes and defective fuel is provided for by this special rack. This rack was analyzed for storage of ruptured fuel as shown in Figure 9.1-21. Special rack input parameters are summarized in Table 9.1-19.

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Insert A page 14 of 16

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The storage of ruptured fuel is a more reactive evaluation than that of control rods or control rod guide tubes.

[redacted] <u>Storage of Ruptured Fuel in the Fully Flooded Special Rack</u>

The storage of ruptured fuel assemblies within defective fuel storage containers inserted into the special ~~6 x 6 un-borated~~ rack was analyzed using the CHEETAH-B/PDQ-7 diffusion theory model. The case was analyzed as an infinite array in order to simulate storage of 30 ~~27 or more~~ ruptured fuel assemblies in the special rack. The resulting K_{eff} for this case was .6589. Considering that this K_{eff} accounts for no radial or axial leakage, the reactivity for the storage of fuel in the special rack is well below the design limit K_{eff} of .95.

Storage of undamaged fuel within the special rack is less reactive than storage of damaged fuel. This is due to the fact that in the ruptured fuel case, the defective fuel storage container displaces water. For this reason, the storage of undamaged fuel was not analyzed.

DSER OPEN ITEM 140

[redacted]

Insert H page 15 of 16

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9.1.2.3.3.6

[redacted] SUMMARY AND CONCLUSION

The final result as calculated by both the working model (CHEETAH-B/CORCBLADE/PDQ-7) and the verification model (AMPX/KENO-IV) is summarized in this section and compared to the NRC regulation k_{eff} limit of 0.950. The "Reference Case" referred to in this report uses the nominal dimensions given in Figure 9.1-20 and Table 9.1-19 without the dimensional and material tolerances included.

[redacted] Results of the Transport Monte Carlo (AMPX/KENO-IV) Verification Calculations and the Calculational Bias

k_{eff}, Reference Case 9.1.2.3.3.1 [redacted]	0.9306 ± 0.0042
Benchmark bias, Δk	-0.001
	.9296 ± 0.0042
95% Confidence Interval k_{eff}	0.9212 - 0.9380

The bias of the KENO-IV vs. measurement is based on criticality experiments performed with fixed neutron poisons [redacted]. These experiments were chosen because they approach the fuel storage rack configuration in that they used fixed poison plates between fuel rod clusters. The result of the benchmark calculations was that the KENO-IV results were 0.001Δk above the measured value. This demonstrates a negative bias of 0.001Δk.

[redacted] Summary of Results

k_{eff}, adjusted (KENO [redacted])	0.9296 ± 0.0042
Dimensional and Positional Tolerance, Δk	
(PDQ [redacted])	0.0037
PDQ correction for non-conservative	
assumptions in the reference case,	
Δk (PDQ [redacted])	0.0006
Dropped Assembly, Δk (PDQ [redacted])	0.0063

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

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

Insert A page 16 of 16

HCGS FSAR

Final K_{eff}	0.9402 ± 0.0042
95% Confidence Interval	0.9318 - 0.9486
Design Limit, k_{eff}	~~0.9318 - 0.9486~~ 0.950

The final k_{eff} value (0.9486) includes all the design specification tolerances, the postulation of a dropped fuel assembly, the model bias, and the 95% confidence interval from the KENO calculations. However, the negative reactivity effect (~ 0.5% Δk) due to the presence of U-234 and the parasitic structure materials (i.e., spacer grids) in each assembly was not included.

gates, and any other nonroutine heavy loads that must be carried over the spent fuel pool.

9.1.6 REFERENCES

9.1-1 C. L. Martin, Lattice Physics Method, NEDO-20913, General Electric, June 1975.

9.1-2 AISC Manual of Steel Construction

9.1-3 AGMA Gear Classification Manual

9.1-4 Aluminum Construction Manual, Aluminum Association

9.1-5 AWS D1.1, Structural Welding

9.1-6 NEMA MG-1, Motor and General Standards

9.1-7 National Electric Code

9.1-8 OSHA 1910.179

9.1-9 OSHA, Vol 37, No. 202, Part 191 ON

9.1-10 ANSI N 210-1976, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations."

9.1-11 UAI 84-42 Revision 0, Summary Report of Nuclear Criticality Analysis For the Spent Fuel Racks of Hope Creek Generating Station.

CRITICALITY REFERENCE CASE SPENT FUEL RACK INPUT PARAMETERS

FUEL ASSEMBLY (8x8)

FUEL DATA

Pellet OD	0.410"
Clad OD	0.483"
Clad Thickness	0.032"
Clad Material	Zr-2
Fuel Rod Pitch	0.640"
Active Fuel Length	150."
U-235 Enrichment	3.4 w/o
Effective (Stacked) Density	96.5% Theoretical

WATER ROD DATA (2 per Assembly)

Water Rod OD	0.591"
Water Rod Thickness	0.030"
Water Rod Material	Zr-2

CHANNEL DATA

Channel Inside Dimension	5.27"
Channel Thickness	.080"
Channel Material	Zr-4

TABLE 9.1-19 (continued)

BORAL PLATE DATA

Total Thickness	0.095" + .005" - .010
Width	4.465" ± .0625"
Length	135" ± 0.25"
Sheath Thickness	0.011" ± .001"
Sheath Material	Aluminum (1100 series)
Core Thickness	0.073" (nominal; range: .061" to .080
Core Material	Boral
B-10 Density	0.028 g/cm^2

CAVITY DATA

Can Inside Dimension	6.080" (nominal)
Can Thickness	0.090" (nominal)
Outer Wrapper Inside Width Dimension	4.562" ± .020"
Outer Wrapper Outside Width Dimension	5.36" (nominal)
Outer Wrapper Inside Thickness Dimension	0.101" (nominal)
Outer Wrapper Material Thickness	0.024" (nominal)
Cavity Material	Stainless Steel
Rack Cavity Pitch	6.308" ± 0.030"

TABLE 9.1-20

CRITICALITY [redacted] REFERENCE CASE - SUMMARY OF 4-GROUP PDQ-7 RESULTS

k_{eff} AS A FUNCTION OF TEMPERATURE AND VOIDS

REFERENCE GEOMETRY - FIGURE 9.1-20

Temperature °F	% Voids	K_{eff}
40	0	0.9265
68	0	0.9235
212	0	0.9028
212	10	0.8845
212	20	0.8630

TABLE 9.1-20

[redacted]

k_{eff} AS A FUNCTION OF PITCH

VARIATION (INCHES)	PITCH (INCHES)	K_{eff}
+ .030	6.338	.9220
Base	6.308	.9235
- .030	6.278	.9250
- .060	6.248	.9264

TABLE 9.1-20

k_{eff} AS A FUNCTION OF BORAL CORE THICKNESS - AREAL DENSITY CONSTANT

	Thickness of Boral Core Inches	K_{eff}
	0.061	0.9233
(base)	0.073	0.9235
	0.080	0.9236

HCGS FSAR

TABLE 9.1-21

[redacted] SPECIAL NON-POISONED SPENT FUEL RACK INPUT PARAMETERS FOR THE CRITICALITY ANALYSIS

FUEL ASSEMBLY (8x8)

FUEL DATA

Pellet OD	0.410"
Clad OD	0.483"
Clad Thickness	0.032"
Clad Material	Zr-2
Fuel Rod Pitch	0.640"
Active Fuel Length	150."
U-235 Enrichment	3.4 w/o
Effective (Stacked) Density	96.5% Theoretical

WATER ROD DATA (2 per Assembly)

Water Rod OD	0.591"
Water Rod Thickness	0.030"
Water Rod Material	Zr-2

CHANNEL DATA

Channel Inside Dimension	5.27"
Channel Thickness	.080"
Channel Material	Zr-4

CAVITY DATA

Can Inside Dimension	11.50 + .00" - .06"
Can Thickness	0.165 (nominal)
Cavity Material	Stainless Steel
Rack Cavity Pitch	11.663" + .020" - .000"

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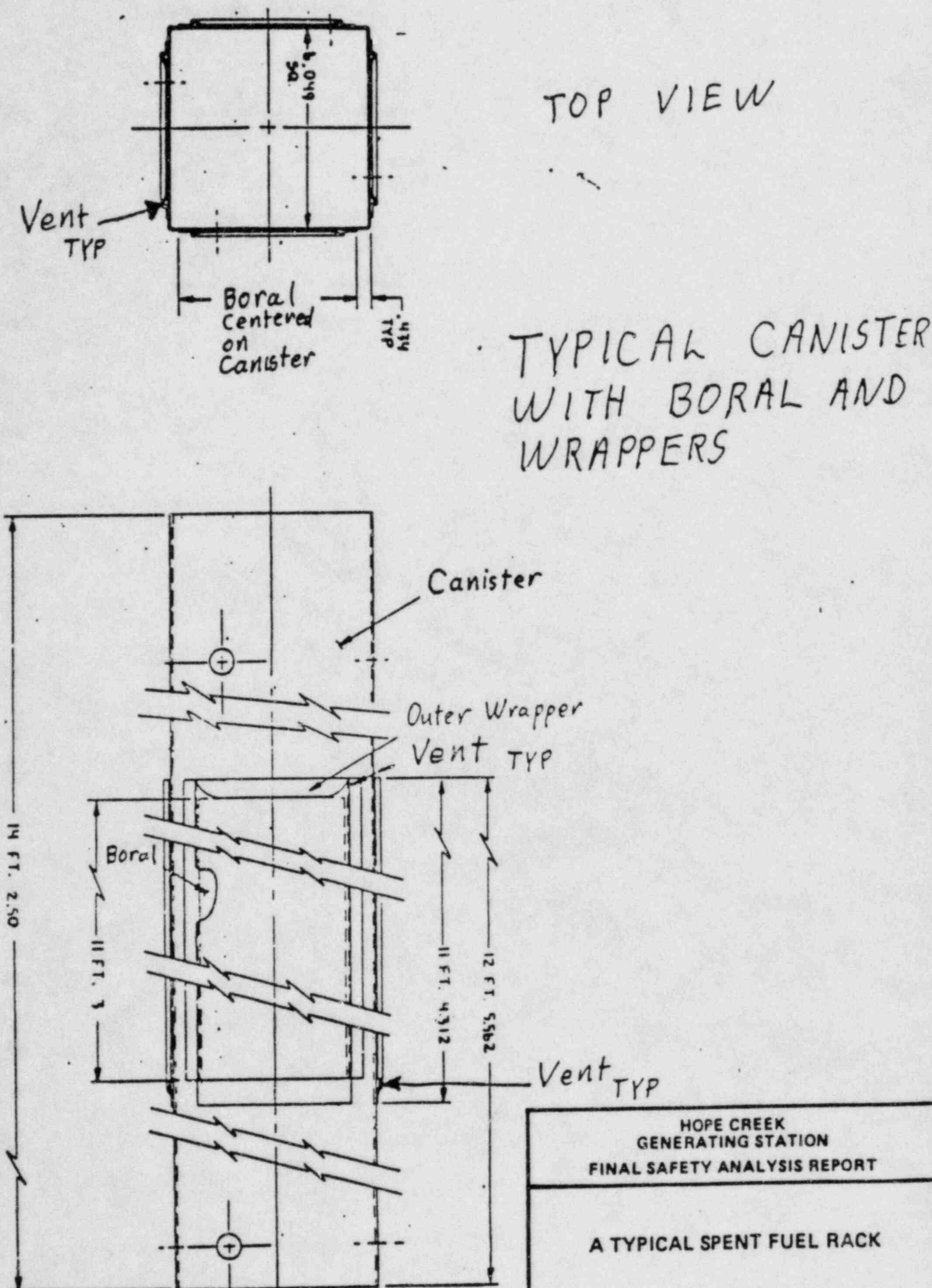

HOPE CREEK
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FINAL SAFETY ANALYSIS REPORT

A TYPICAL SPENT FUEL RACK

FIGURE 9.1-3 Sheet 4 of 4.

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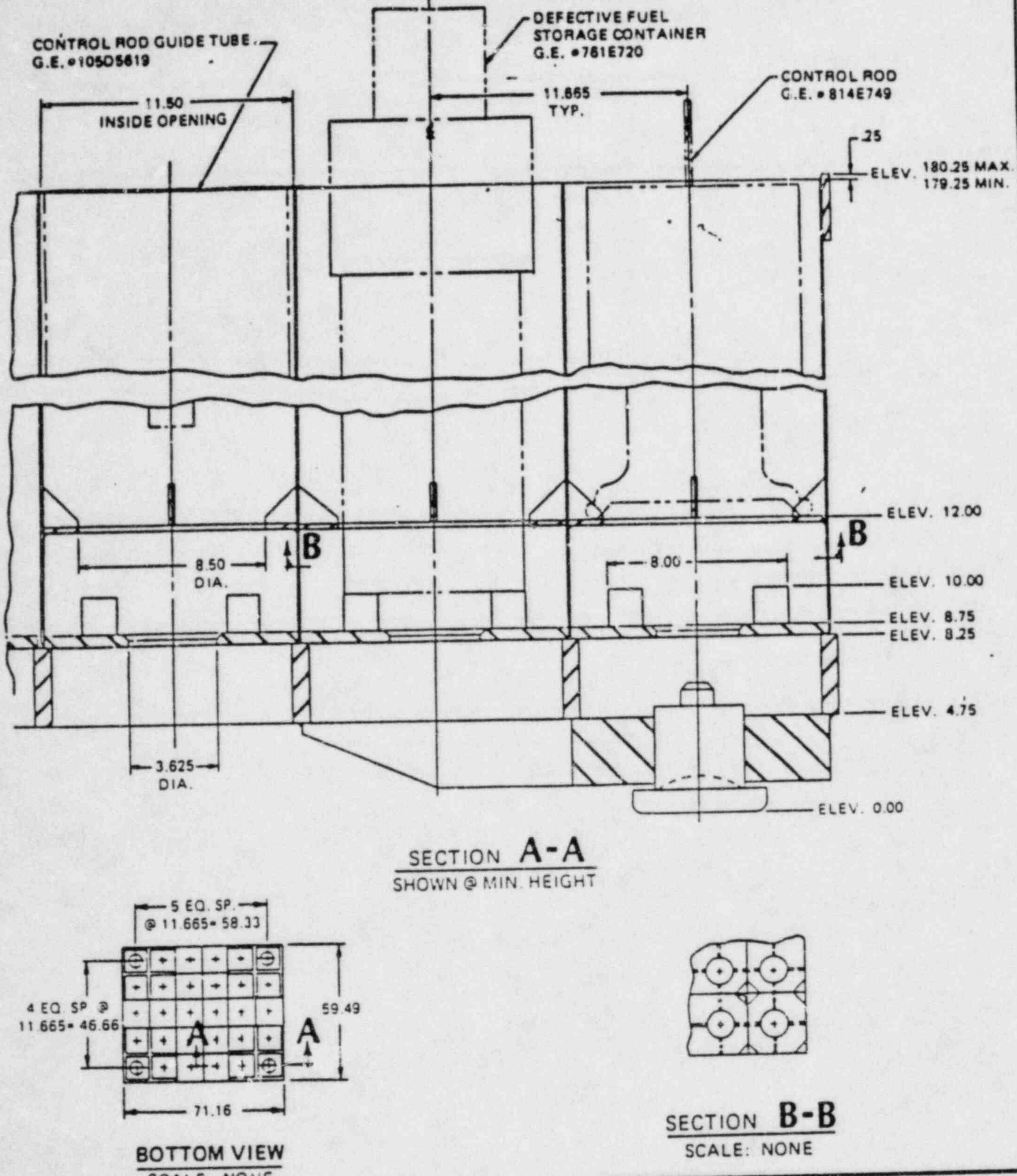

NOTES:

1. REGULAR FUEL BUNDLES COULD ALSO BE STORED IN THIS RACK ON A TEMPORARY BASIS IF NECESSARY.

2. 5 x 6 SPECIAL RACK FOR STORAGE OF CONTROL ROD G.E. #814E749 AND DEFECTIVE FUEL STORAGE CONTAINER G.E. #761E720 AND CONTROL ROD GUIDE TUBE G.E. #105D5619.

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FINAL SAFETY ANALYSIS REPORT

SPECIAL
A ~~TYPICAL~~ SPENT FUEL RACK

FIGURE 9.1-~~5~~ 19 sheet 1 of 2
~~SHEET 4 OF 4~~ ~~AMENDMENT 7, 08/84~~

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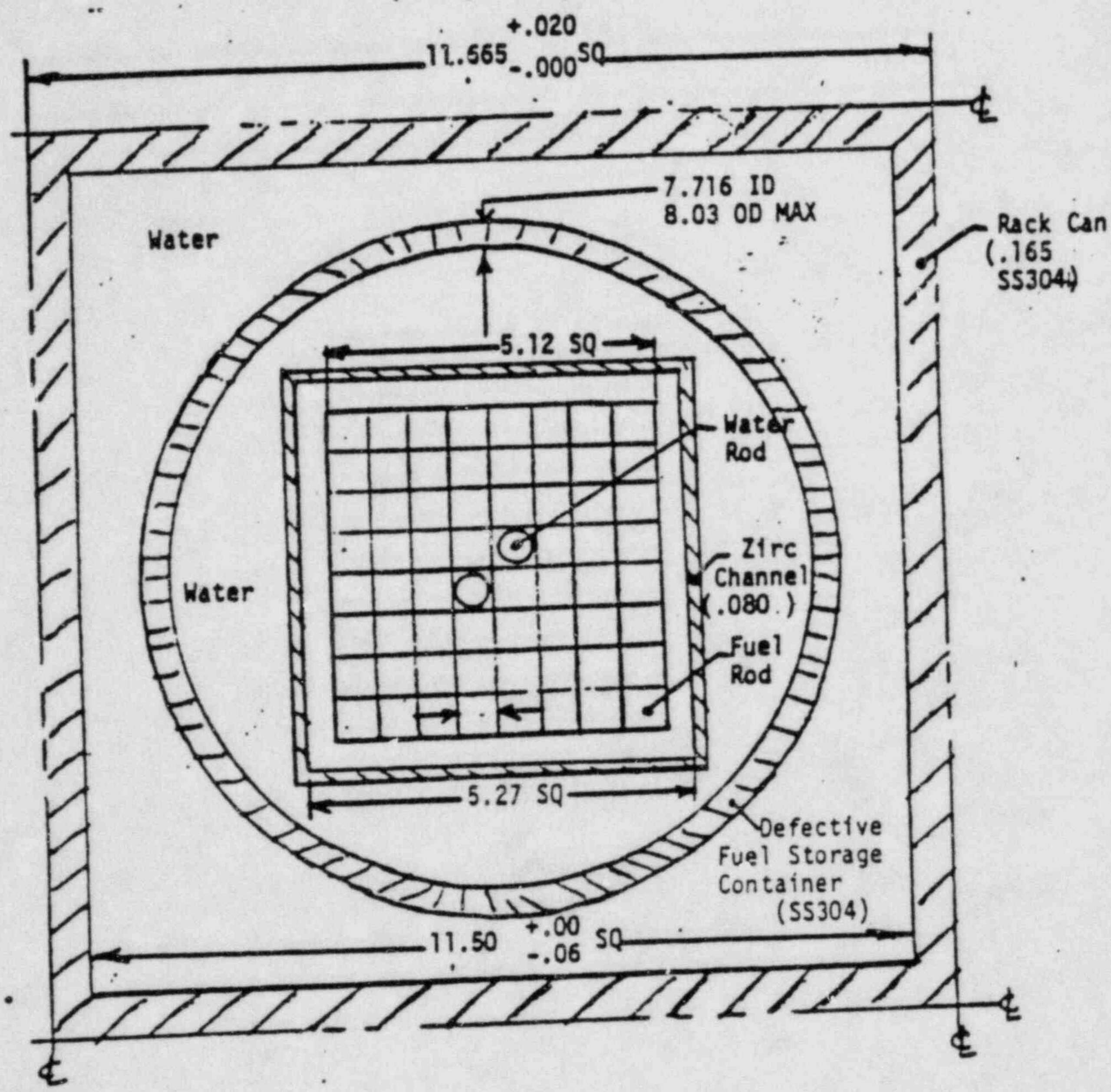
~~FIGURE 19.~~ SPECIAL ~~8x8 UNPOISONED~~ RACK



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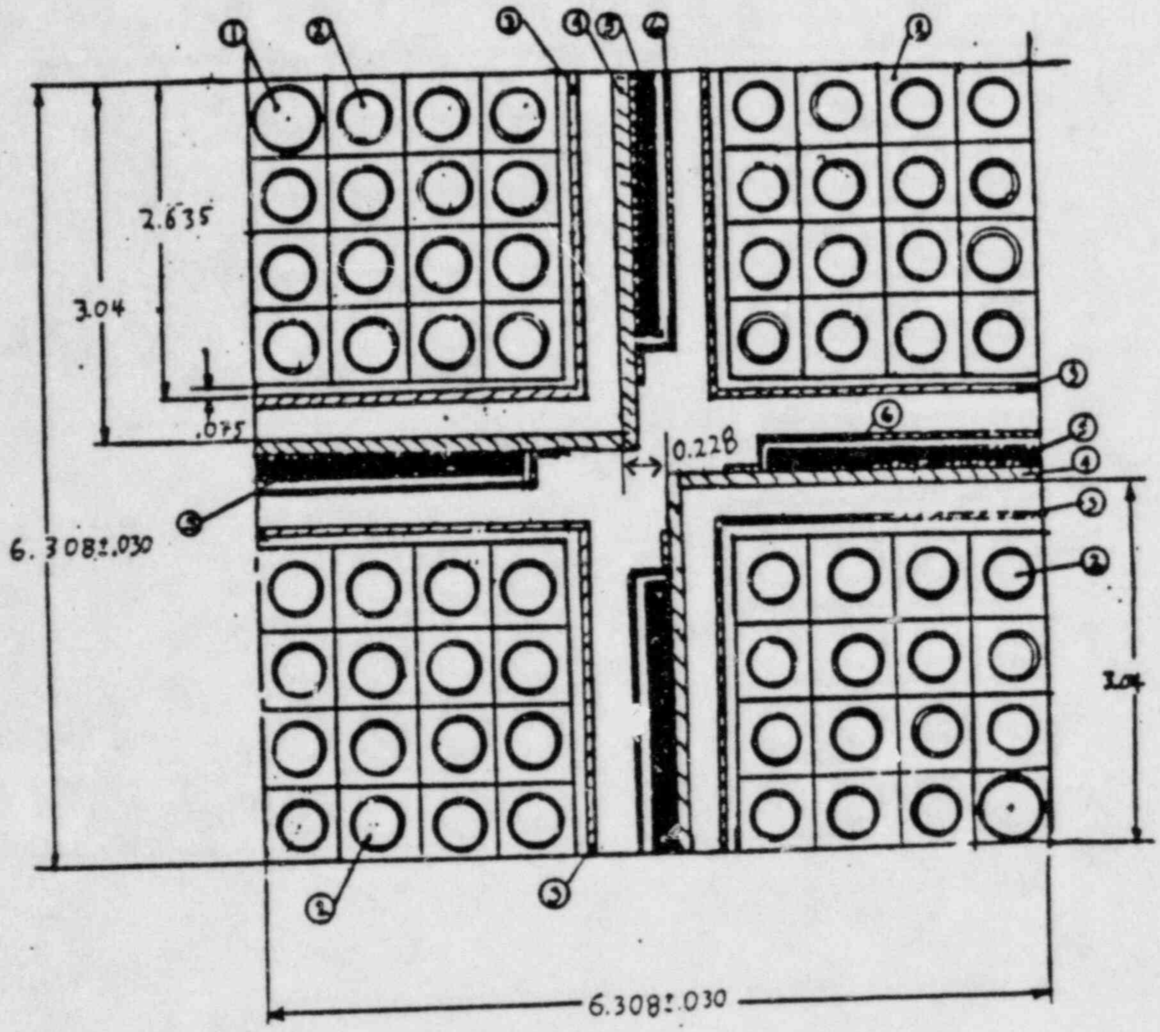
A SPECIAL SPENT FUEL

Figure 9.1-19 sheet 2

HOPE CREEK SPENT FUEL RACK
REFERENCE CASE GEOMETRY



NOTE:
1. All Dimensions are Inches.
2. Refer to Table 9.1-19 for the Parameters and Dimensions of These Items.

#	ITEM	MATERIAL
1	WATER ROD CELL	ZIRC + WATER
2	FUEL ROD CELL	ZIRC, WATER, UO_2
3	CHANNEL	ZIRC
4	INNER CAN	SS304L
5	BORAL	BORAL+AL
6	OUTER WRAPPER	SS304L

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HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

SPENT FUEL RACK
CRITICALITY GEOMETRY

FIGURE 9.1-~~1~~20 Sheet 1 of 2

~~FIGURE 1.~~ ~~HOPE CREEK HIGH DENSITY~~ SPENT FUEL STORAGE RACK, DROPPED ASSEMBLY ACCIDENT

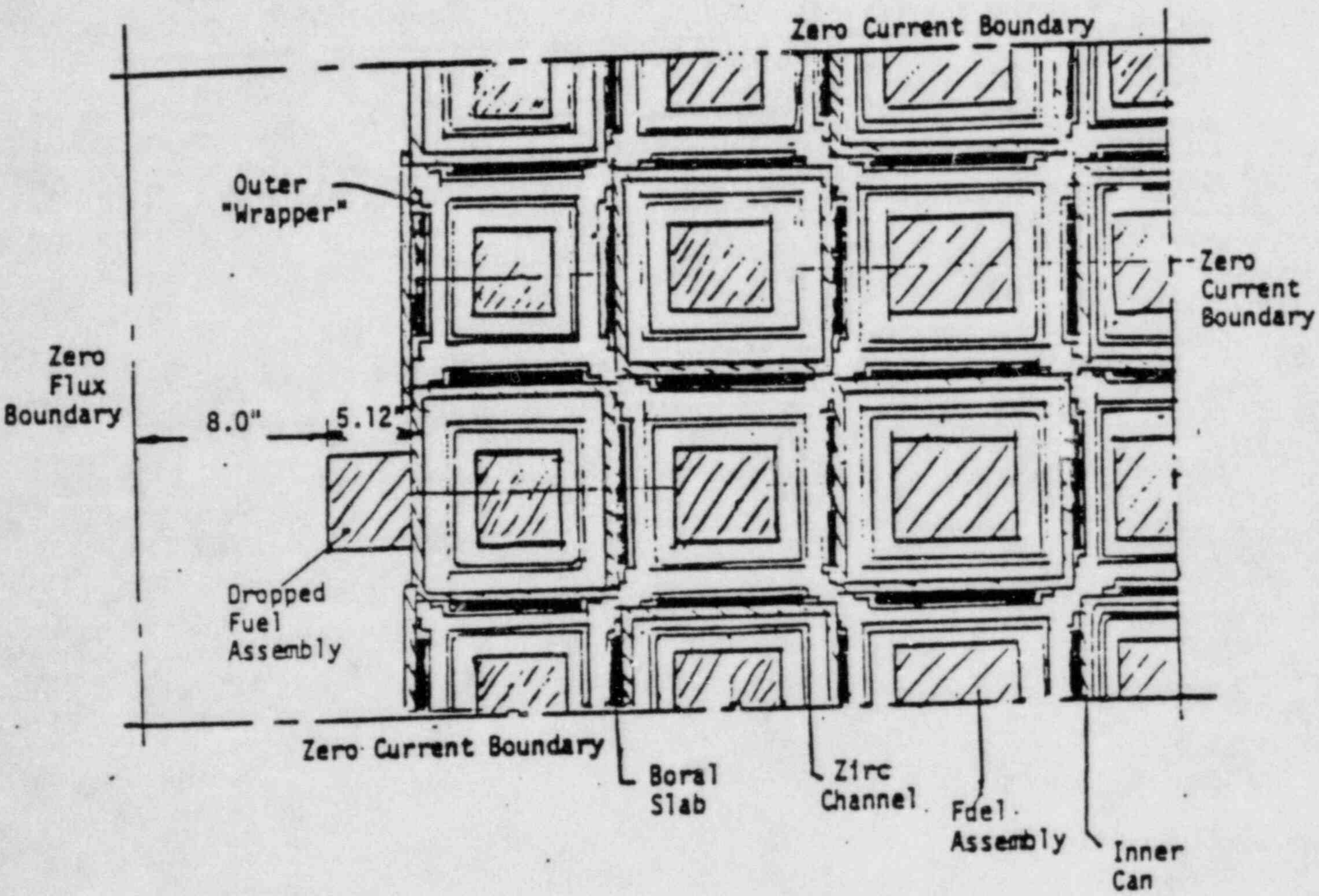

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Figure 9.1-20 sheet 2 of.

APPENDIX 9B

DESIGN, ANALYSIS AND CONSTRUCTION OF ~~HIGH DENSITY~~ SPENT FUEL STORAGE RACKS

9B.1 SCOPE

This appendix describes the design, analysis and construction of the spent fuel racks.

9B.2 DESCRIPTION OF SPENT FUEL POOL AND RACKS

Section 9.1.2.2 contains a description of the spent fuel storage facility including the high density spent fuel storage racks. The spent fuel racks are of free standing design and are not attached to either the fuel pool wall or the fuel pool liner plate. Figures 1.2-10 and 1.2-32 show the spent fuel pool in relation to other plant structures. Figures 9.1.3 and 9.1.4 show details of the spent fuel racks.

The spent fuel racks are designed to withstand the postulated drop of a fuel bundle. Section 9.1.5 contains a description of the overhead heavy load handling systems for the reactor building polar crane including figures showing load paths for the crane.

9B.3 APPLICABLE CODES, STANDARDS AND SPECIFICATIONS

All parts of the spent fuel racks, except the adjusting screws in the feet of each module and the poison material, are made from ASTM A240, Type 304L stainless steel. The adjusting screws are made from ASTM A564, Type 630 stainless steel. Boral is the poison material.

Design, fabrication and installation of the spent fuel racks are performed based upon Subsection NF requirements of Reference 9B-1 for Class 3 component supports.

9B.4 SEISMIC AND IMPACT LOADS

The seismic input for the spent fuel racks consists of floor response spectra for the spent fuel pool slab. Floor response spectra are developed from ground response spectra which comply with the requirements of Regulatory Guides 1.60 and 1.61. Acceleration time histories are developed for two horizontal directions and one vertical direction from the floor response spectra. These three time histories are imposed simultaneously. The peak responses from each direction are combined by square root of the sum of the squares in accordance with Regulatory Guide 1.92.

Impact loads due to fuel rattling are calculated using methods described in Section 9B-6. Impact loads are considered for local as well as overall effects on the rack design.

9B.5 LOADS AND LOAD COMBINATIONS

Loads and load combinations are in agreement with Table 1 of Reference 9B-2. Thermal effects are included by using decreased material properties at the applicable temperature level. Since the racks are free standing, there are no thermal stresses.

9B.6 DESIGN AND ANALYSIS PROCEDURES

Each fuel rack is idealized as a 3D finite element model using the ANSYS computer program. Figure 9B-1 shows a five canister portion of a rack. The canisters and bottom grid plate are modeled with plate elements. The perimeter bar, which secures the canisters at the top, and the stiffening bars for the grid plate are modeled with beam elements. The thin stainless steel wrapper containing the neutron absorber and the stainless steel panels used to close off the alternate cavities are not modeled but their masses are included. The fuel assemblies are modeled as beam elements.

Figure 9B-2 shows a double rack model in schematic form. 3D interface elements are used to represent the fuel-to-canister clearance as well as the rack-to-rack gap. These nonlinear elements reproduce forces due to fuel rattling and possible rack-to-rack interaction. 3D gap elements with material properties based on the interface friction coefficients are used to simulate the corner supporting feet which may slide or lift off the pool floor. Two bounding values of friction coefficient (0.2 and 0.8) are used in order to identify the most critical conditions for sliding and for maximum reactions at the support feet.

Structural damping coefficients of 2 per cent for OBE and 4 per cent for SSE are used, except that impact damping of 10 per cent of critical is used for the gap elements since impact dissipates substantial amounts of energy. With 20 feet of submergence, sloshing effects are negligible and therefore are neglected. Fluid damping effects are also neglected. To simulate the immersion effects, all the internal water entrapped within the rack envelope is added to the horizontal mass. The external water between adjacent racks is modeled using the hydrodynamic coupling element shown in Figure 9A-2.

A parametric study, which considers varying amounts of fuel in a single rack, is conducted to determine which of the following conditions should be considered in order to maximize the seismic response of the racks.

- rack empty
- rack one-third full

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- rack two-thirds full
- rack full

For the partially loaded conditions, eccentricity of the fuel on one side of the rack is considered.

9B.7 STRUCTURAL ACCEPTANCE CRITERIA

Allowable stresses are in agreement with Table 1 of Reference 9B-2. Stress levels for beam elements comply with the requirements of Appendix XVII to Reference 9A-1. Stress levels for plate elements comply with rules for plate and shell type supports since stress fields in these components are biaxial.

For the load drop condition, local permanent deformation possibly requiring repair is permissible provided that overall stresses do not exceed values permitted for Level D service limits and the resulting deformation does not permit the fuel configuration K_{eff} to exceed 0.95.

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9B.8 MATERIALS, QUALITY CONTROL AND SPECIAL CONSTRUCTION TECHNIQUES

Materials are described in Section 9B.3. Quality control procedures for materials, fabrication and design control and verification comply with ANSI N45.2. Conventional construction methods are used.

As described in Section 9.1.2.2.2.2, approximately 25 per cent of the total spent fuel storage capacity will be provided by racks installed prior to initial plant operation. The remaining racks will be installed later. The initially installed racks are generally located at the north end of the spent fuel pool. Therefore, the additional racks can be installed later without being transported over existing racks which contain spent fuel.

9B.9 REFERENCES

9B-1 ASME Boiler and Pressure Vessel Code, Section III, Division 1, 1980 Edition, Summer 1982 Addenda.

9B-2 NRC NUREG-0800, SRP Section 3.8.4, Appendix D, Rev. 0, July 1981.

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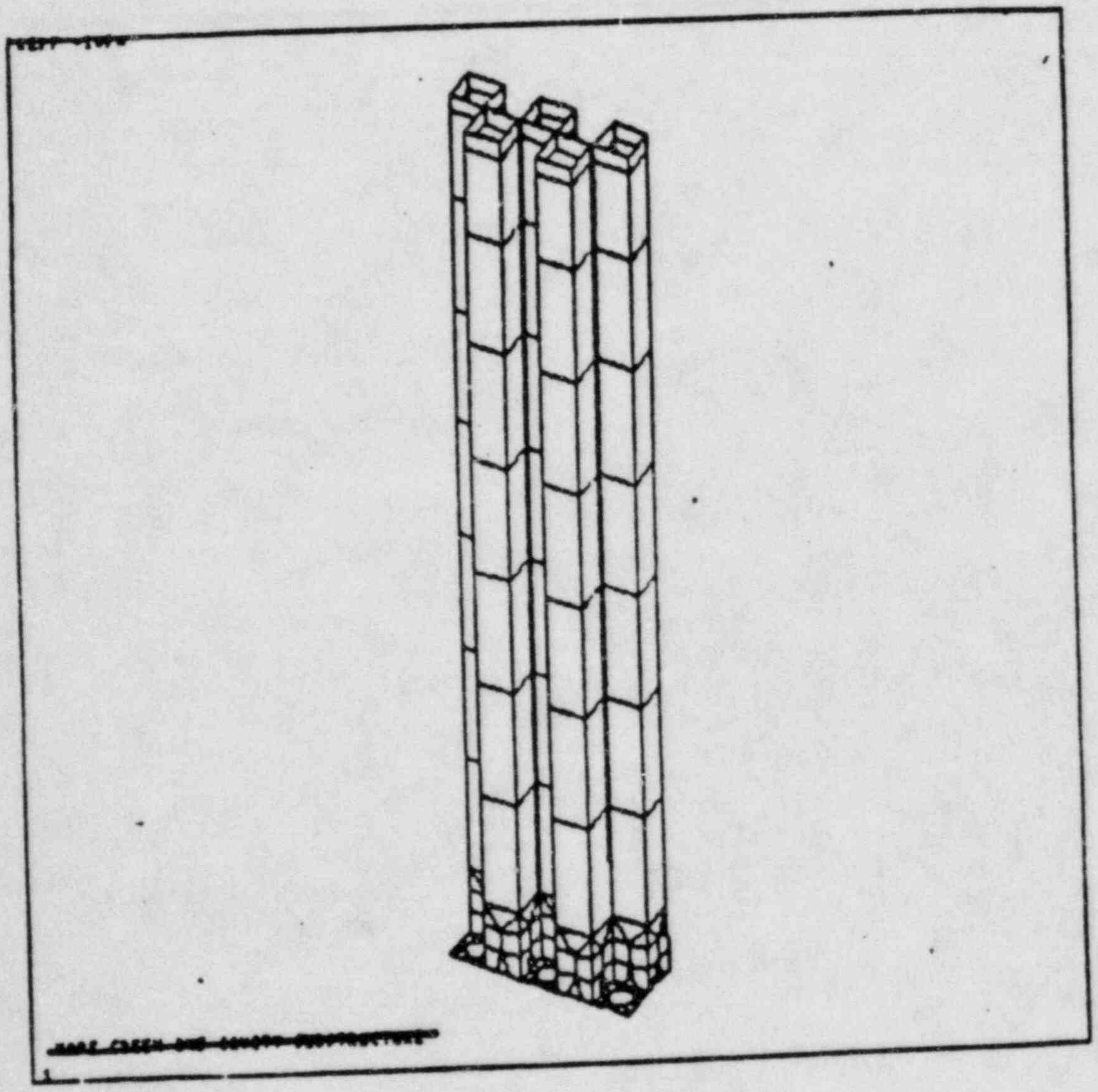

Figure 9B-1 Model for canister and portion of bottom grid

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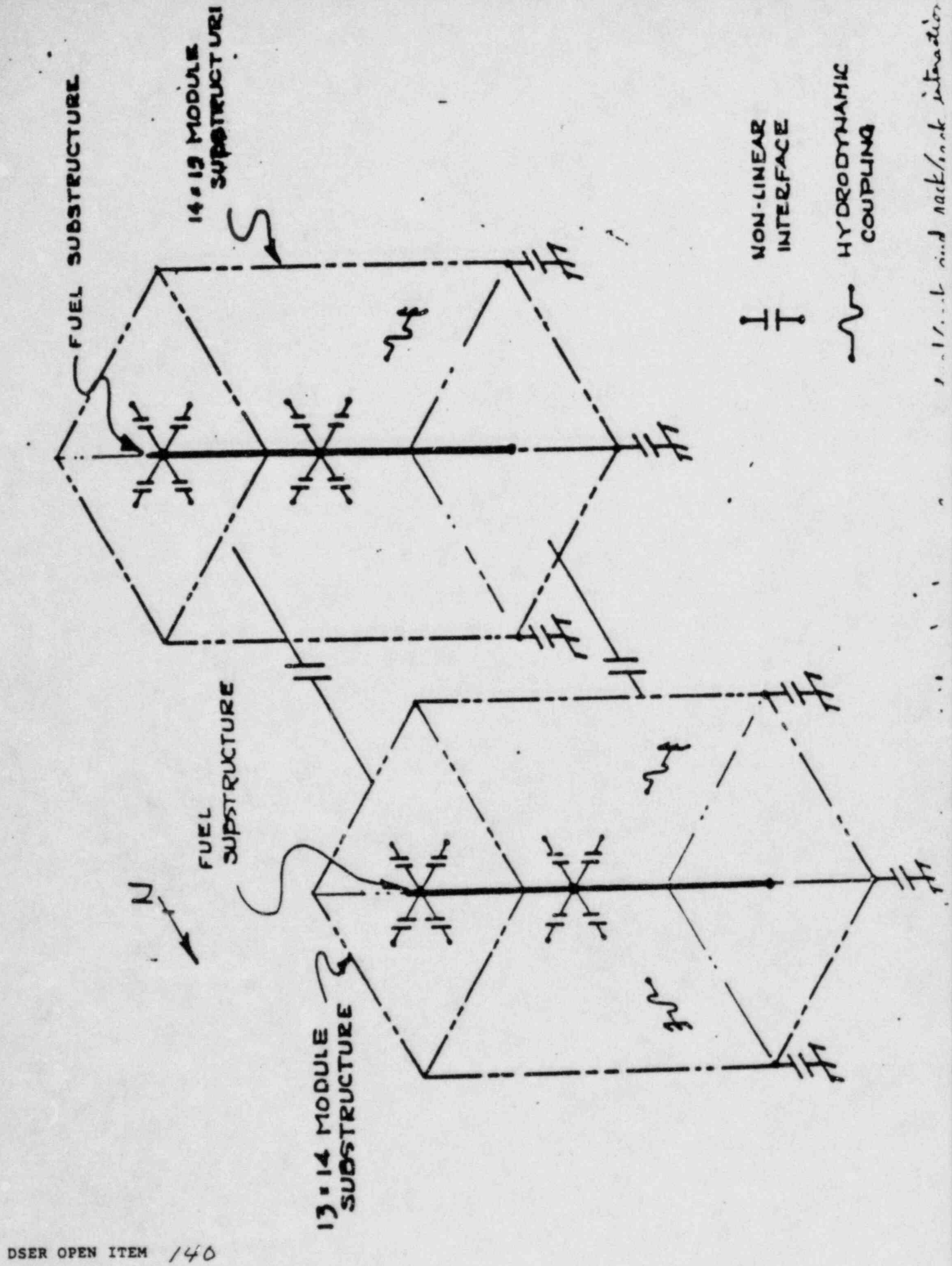

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QUESTION 220.15 (SECTION 3.8.4)

Provide sketches of the mathematical models used in the design of spent fuel racks. Describe in detail, the methods of analysis by which seismic and other loads are applied to the racks and the pool.

RESPONSE

~~The requested information will be available by June, 1984, and will be added to Section 3.8.4 and/or 9.1.2 as appropriate.~~

Sections ~~3.8.4.4.1~~ 3.8.4.8.3 and 9.1.2.2.2.2 have been revised and Appendix 9B has been added to provide the requested information.

QUESTION 281.13 (SECTION 9.1.2)

Identify the materials, including the neutron absorbing material (poison), used in the fabrication of the high density spent fuel storage racks and all other structural components wetted by the pool water. Indicate how the poison-containing cavities are vented.

RESPONSE

All parts of the spent fuel racks, except the adjusting screws in the feet of each module and the poison material, are made from ASTM A240, Type 304L, stainless steel. The adjusting screws are made from ASTM A564, Type 630 stainless steel with H 1100 heat treatment. Heat treatment scale removed. Boral is the poison material.

Thin (0.024 inch thick) outer canister sheets hold the Boral tighltly against the 0.090 inch thick inner canister walls. The outer canisters are spot welded to the inner canisters along the bottom and both vertical sides of the outer canister. The top edge of each outer canister is seam welded to the inner canister. ~~The gaps between the spot welds provide~~ The poison vents ~~ing~~ are shown on Figure 9.1-3.

QUESTION 281.14 (SECTION 9.1.2)

Provide details of the materials monitoring program for the spent fuel pool, including type of samples used and frequency of inspection.

RESPONSE

There are no plans to provide a materials monitoring program at HCGS, as vented stainless steel spent fuel racks with Boral poison material are already in use at other BWR spent fuel storage pools, such as Monticello and Browns Ferry. PSEG will develop a program to monitor the Boral surveillance program of either Fermi, Monticello or Brown's Ferry by March 1985. However, forty-five test coupons have been installed in high radiation areas of the spent fuel pool in case a need for a HCGS materials monitoring program is indicated by the Boral surveillance program.

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QUESTION 410.38 (SECTION 9.1.2)

Insufficient information is provided for review of the criticality of the spent fuel pool. The design bases are acceptable with respect to criticality. The information required for the review is promised for later. Such information should include the following:

a. Sufficient structural detail to permit an independent calculation of the criticality of the racks.

b. A description of the calculational methods used along with the results of the verification of the methods. This may be by reference to documents previously submitted by the organizations doing the analysis.

c. A tabulation of the nominal value of k effective of the racks along with the various uncertainties and biases considered in the analysis.

d. A tabulation of the reactivity effect of each of the abnormal (accident) situations considered.

RESPONSE

~~Sufficient information for review of the criticality of the spent fuel pool, including that listed above will be available by September 1984, and will be added to Section 9.1.2.~~

Section 9.1.2.3.3 has been revised to include the information requested above

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AAR BROOKS & PERKINS CORP.
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Report 624

Boral
Neutron Absorbing/Shielding Material

Product Performance Report

Prepared By:

L. Mollon
Nuclear Programs Manager
July 20, 1982

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TABLE OF CONTENTS

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BORAL
NEUTRON ABSORBING/SHIELDING MATERIAL

Product Performance Report

GENERAL

Boral is a thermal neutron poison material composed of boron carbide and the 1100 alloy aluminum. Boron carbide is a compound having a high boron content in a physically stable and chemically inert form. The 1100 alloy aluminum is a light-weight metal with high tensile strength which is protected from corrosion by a highly resistant oxide film. The two materials, boron carbide and aluminum, are chemically compatible and ideally suited together for long-term use in the radiation environment of a nuclear reactor or in spent fuel containment.

Boral is an ideal neutron absorbing/shielding material because of the following reasons:

1. The content and placement of boron carbide provides a very high removal cross section for thermal neutrons.
2. Boron carbide, in the form of fine particles, is homogenously dispersed throughout the central layer of the Boral panels.
3. The boron carbide and aluminum materials in Boral are totally unaffected by long-term exposure to gamma radiation.
4. The neutron absorbing central layer of Boral is protected by permanently attached surfaces of aluminum.

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5. Boral is stable, strong, durable, and corrosion resistant.

Boral is manufactured under the control and surveillance of a computer-aided Quality Assurance/Quality Control Program that conforms to the requirements of 10CFR50 Appendix B entitled, "Quality Assurance Criteria for Nuclear Power Plants". For further discussion on Quality Control see Brooks & Perkins Bulletin No. 102.

Boral has been licensed by the USNRC for use in BWR and PWR spent fuel storage racks, shipping and storage containers and for many other shielding uses including control blades. For specific applications see later in this report.

Boral panels can be used in the flat panel form or fabricated into a variety of geometrical shapes by standard metal working methods and techniques. The shielding capability of Boral is assured by wet chemical analysis or neutron attenuation testing and is specified as a minimum of grams of B^{10} per square centimeter of surface area. Boral can be provided at any B^{10} loading up to 0.06 gm/sq cm as required.

BORAL MATERIAL CHARACTERISTICS

Aluminum. Aluminum is a silvery-white, ductile metallic element that is the most abundant in the earth's crust. The 1100 alloy aluminum is used extensively in cooking utensils, heat exchangers, pressure and storage tanks, chemical equipment, reflectors and sheet metal work.

It has high resistance to corrosion in rural, industrial and marine atmospheres. Aluminum has atomic number of 13, atomic weight of 26.98, specific gravity of 2.69 and valence of 3. The physical and mechanical properties of the 1100 alloy aluminum are listed in Table 1.

Table 1 - 1100 Alloy Aluminum

Density	0.098	lb/cu. in.
	2.713	gm/cc
Melting Range	1190-1215	deg. F
	643-657	deg. C
Thermal Conductivity	128	BTU/hr/sq ft/deg. F/ft
(77 deg. F)	0.53	cal/sec/sq cm/deg. C/cm
Coef. of Thermal Expansion	13.1×10^{-6}	/deg. F
(68-212 deg. F)	23.6×10^{-6}	/deg. C
Specific Heat	0.22	BTU/lb/deg. F
(212 deg. F)	0.23	cal/gm/deg. C
Modulus of Elasticity	10×10^{6}	psi
Tensile Strength	13,000	psi annealed
(75 deg. F)		
	18,000	psi as rolled
Yield Strength	5,000	psi annealed
(75 deg. F)		
	17,000	psi as rolled
Elongation	35-45%	annealed
(75 deg. F)		
	9-20%	as rolled
Hardness	23	annealed
(Brinell)		
	32	as rolled
Annealing Temperature	650	deg. F
	343	deg. C

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Chemical Composition - Aluminum (1100 Alloy)

99.00% min. - Aluminum

1.00% max. - Silicon and Iron

.05-.20% max. - Copper

.05% max. - Manganese

.10% max. - Zinc

.15% max. - others each

The excellent corrosion resistance of the 1100 alloy aluminum is provided by the protective oxide film that develops on its surface from exposure to the atmosphere or water. This film prevents the loss of metal from general corrosion or pitting corrosion and the film remains stable between a pH range of 4.5 to 8.5. More detailed corrosion data is provided later in this report and in Brooks & Perkins Bulletin No. 101.

Boron Carbide. The boron carbide contained in Boral is a fine granulated powder that conforms to ASTM C-750-80 nuclear grade Type III. The particles range in size between 60 and 200 mesh and the material conforms to the chemical composition listed in Table 2.

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Table 2 - Boron Carbide Chemical Composition, Weight %

Total boron	70.0 min.
B^{10} isotopic content in natural boron	18.0 min.
Boric oxide	3.0 max.
Iron	2.0 max.
Total boron plus total carbon	94.0 min.

The general physical properties of the boron carbide powder are listed in Table 3.

Table 3 - Boron Carbide Physical Properties

Chemical formula	B_4C
Boron content (weight)	78.28%
Carbon content (weight)	21.72%
Crystal structure	rombohedral
Density	2.51 gm/cc-0.0907 lb/cu. in.
Melting point	2450°C-4442°F
Boiling point	3500°C-6332°F
Microscopic capture cross section	600 barn

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Materials Compatibility. The materials contained in Boral are compatible with all parts of a spent fuel storage system in either a boiling-water (BWR) or pressurized-water reactor including the fuel assemblies, the cooling system, the cleanup system, the pool liner and the structures of the storage racks. This compatibility is evidenced by more than seventeen years of continuour service in both types of pool water (1) (3). None of the following materials are contained in Boral nor do they come in contact with Boral during its manufacture and therefore Boral can not cause these materials to come in contact with the fuel assemblies:

a. Any material that contains halogens in amounts exceeding 50 ppm, including chlorinated cleaning compounds.
b. Lead
c. Mercury
d. Sulfur
e. Phosphorus
f. Zinc
g. Copper and Copper alloys
h. Cadmium
i. Tin
j. Antimony
k. Bismuth
l. Mischmetal
m. Carbon steel, e.g., wire brushes
n. Magnesium oxide, e.g., insulation
o. Neoprene or other similar gasket materials made of halogen-containing elastomers.
p. Viton
q. Saran
r. Silastic Ls-53
s. Rubber-bonded asbestos
t. TFE (Teflon) containing more than 0.075% total chlorine (glass-filled) and TFE films containing more than 0.05% total chlorine.

u. Nylon containing more than 0.07% total chlorine.
v. Polyethylene film (colored) with pigments over 50 ppm fluorine, measurable amounts of mercury or halogens, or more than 0.05% lead.
w. Grinding wheels that have been used on other than stainless steel or Inconel material.
x. Water containing more than 25 ppm halogens during any cleaning operation.
y. Any material that forms alloys or deposits on the fuel assembly.

BORAL PHYSICAL CHARACTERISTICS

Boral is a clad composite of aluminum and boron carbide. The Boral panel consists of three distinct layers. The outer protective layers are solid 1100 alloy aluminum. The central layer contains a uniform aggregate of fine boron carbide particles tightly held within an aluminum alloy matrix. The boron carbide particle in the central layer averages 85 micons in diameter. The average spacial separation is 1.25 to 1.50 particle diameters. The overall thickness of the three layers will vary depending on the B^{10} content in accordance with Figure 1.

The physical characteriestics of a Boral panel will vary of course, according to clad thickness, overall thickness and B^{10} content. A typical Boral panel for spent fuel storage can be described as having 0.020 grams of B^{10} per sq. cm with an overall thickness of $.075 \pm .004$ inches including a nominal clad of .0095 inches on each side. The physical characteristics for that typical panel is as shown in Table 4.

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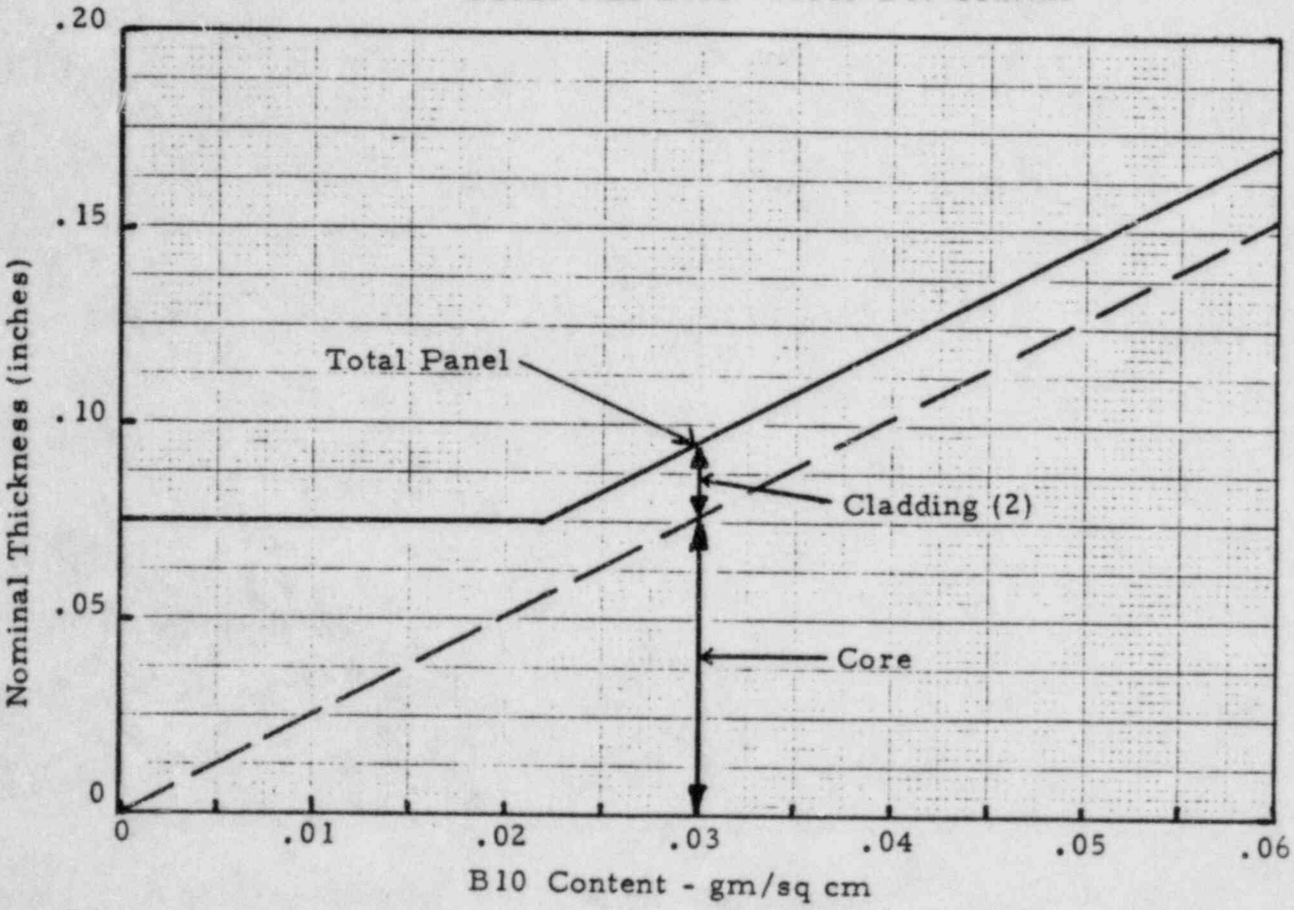
Figure 1

Boral Thickness Versus B10 Content



B10 Content	Equiv. Boron	Total Thickness Including Cladding			
		Inches	± Tol.	mm	± Tol.
.005	.028	.075	.004	1.91	.10
.010	.056	.075	.004	1.91	.10
.015	.083	.075	.004	1.91	.10
.020	.111	.075	.004	1.91	.10
.025	.139	.083	.004	2.11	.10
.030	.167	.096	.005	2.41	.13
.035	.194	.108	.006	2.74	.15
.040	.222	.121	.006	3.07	.15
.045	.250	.133	.006	3.38	.15
.050	.278	.146	.007	3.71	.18
.055	.306	.158	.007	4.01	.18
.060	.333	.172	.009	4.37	.23

This tabulation is for Boral with thin cladding for use in high density spent fuel racks. Boral with thicker cladding is also available for other applications.

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Table 4 - Boral Panel

B^{10} content	0.020 gm/sq cm
Boron content	.111 gm/sq cm
Thickness-overall	.075 $\pm$ .004 inches .190 $\pm$ .010 cm
Thickness-clad (nominal)	.0095 inches .024 cm
Neutron attenuation (at 0.06 eV)	.935
Total weight	.42 gm/sq cm .86 lb/sq ft.

Dispersion Uniformity. The aluminum and boron carbide ingredients in the central core of the Boral panel are combined in powder form. The methods used to weigh and blend the powders as well as the design and construction of the ingots necessary to produce acceptable Boral panels are patented and proprietary processes of Brooks & Perkins. The manufacturing methods used include a sintering process and hot rolling. The final outcome of the entire manufacturing cycle is Boral panels having boron carbide uniformly dispersed throughout the central core. The amount of boron carbide per unit area is directly related to the panel thickness. The minimum B^{10} content per unit area and the uniformity of dispersion within a panel is verified by wet chemical analysis or neutron attenuation testing. For details of the verification methods see Brooks & Perkins Quality Assurance Procedures BP-11002-QAP and BP-11004-QAP.

The acceptance standards in these procedures are controlled by statistical data to assure the minimum requirements are achieved with 95/95 confidence level. The maximum variation in the manufacturing processes (statistical tolerance interval) over a significantly large sample size has been determined and is utilized in the establishment of acceptance criteria.

CORROSION RESISTANCE

The useful service life of Boral will exceed 40 years when in contact with the storage pool water of either a boiling-water or pressurized-water reactor. This fact is evident through laboratory testing and is further supported by the longest continuous, in-pool, service by Boral over any other thermal neutron shielding material. This excellent corrosion resistance is provided by the protective nature of the aluminum cladding that is an integral facing on the Boral panels. The corrosion of aluminum is negligible in fuel storage pools of either type reactor when the water quality and temperatures are maintained within the normal operating limits as listed in Table 5. The boron content in the Boral will not be reduced below the specified limit during the forty or more years of exposure under those operating conditions.

In order to understand the total corrosion resistance of aluminum within the normal operating conditions of the storage pools a discussion of that resistance must consider all forms of corrosion. A detailed discussion follows for general, galvanic, pitting, crevice, intergranular, and stress forms of corrosion.

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Table 5 - Chemistry of Pool Waters

Reactor type	PWR	BWR
Cooling medium	* D-M water	D-M water
Boron content, ppm	0 to 2000	0
pH range	4.5 to 6.0	6.0 to 7.5
Temp range, °F °C	80 to 140 26 to 60	80 to 125 26 to 52
Conductivity @ 25°C micro mho/cm	1 to 30	1
Chloride ions, ppm, max.	0.15	0.20
Fluoride ions, ppm, max.	0.10	----
Total solids, ppm, max.	1.00	0.50
Heavy metals, ppm, max.	----	0.10
Halogens, ppm, max.	0.15	----

* demineralized water

General Corrosion. General corrosion is a uniform attack of the metal over the entire surfaces exposed to the corrosive media. General corrosion is measured by weight loss or decrease in thickness and is generally expressed in mils per year (mpy). The severity of general corrosion of aluminum depends upon the chemical nature and temperature of the electrolyte and can range from superficial etching and staining to dissolution of the metal.

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Figure 2 shows a potential - pH diagram for aluminum in high purity water at 25°C (77°F). The potential for aluminum coupled with stainless steel and the limits of pH for BWR and PWR pools are shown on the diagram to be well within the passivation domain. The passivated surface of aluminum (hydrated oxide of aluminum) affords protection against corrosion in the domain shown because the coating is insoluble, non-porous and adherent to the surface of the aluminum. The protective surface formed on the aluminum (gibbsite and bayerite) is known to be stable up to 135°C (275°F) [5] and in a pH range of 4.5 to 8.5 [6].

Figure 3 is also a potential-pH diagrams for the aluminum-water system but at 60°C (140°F) which also shows the potential for the aluminum/stainless steel couple and the BWR and PWR limits for pH at this upper limit of temperature.

The ability of aluminum to resist corrosion from the boron ions is evident from the wide useage of aluminum in the handling of borax and in the manufacture of boric acid. (7) Aluminum racks with Boral plates in contact with the 800 ppm max. boron water showed only small amount of pitting but maintained good structural integrity after seventeen years in the pool [1].

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Figure 2

Potential Versus pH Diagram

For Aluminum-Water System
At 25°C (77°F) (10)

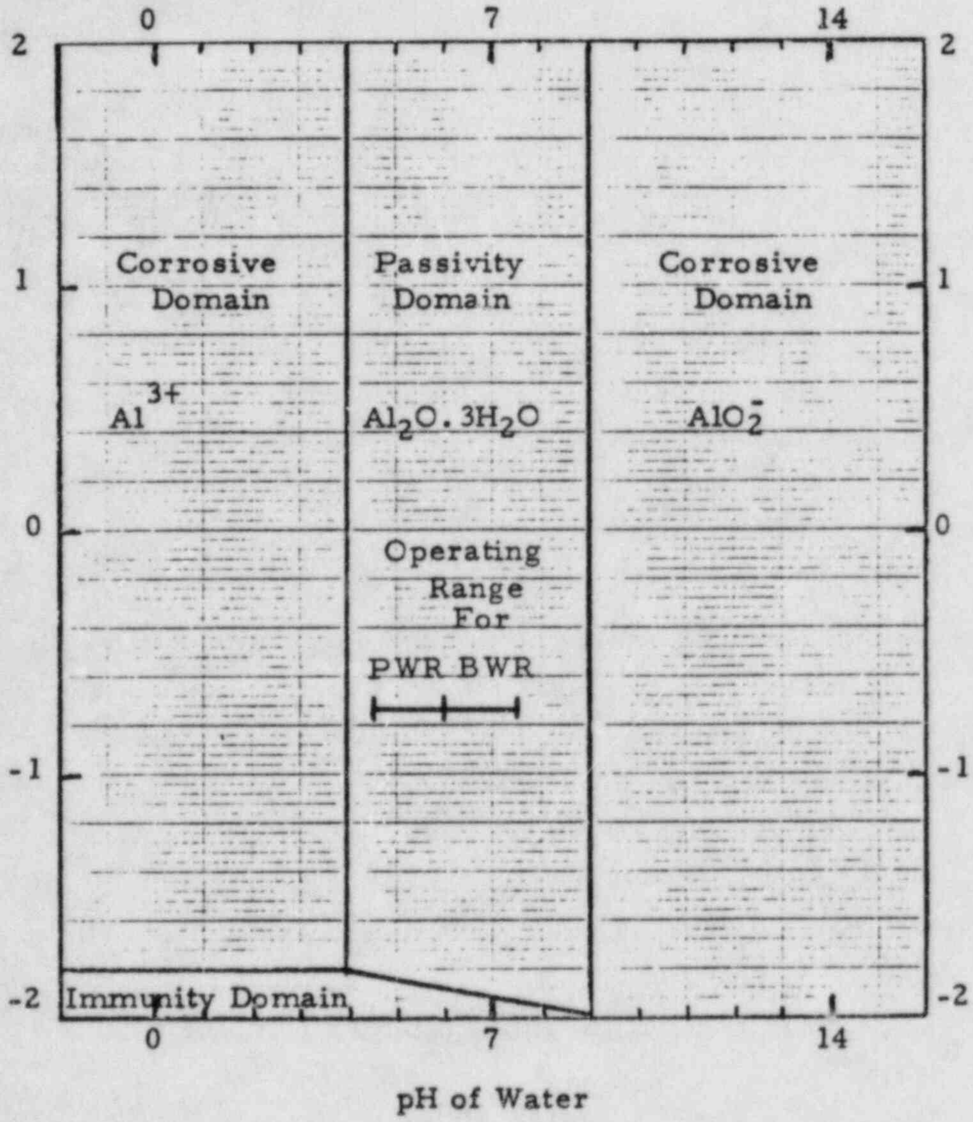

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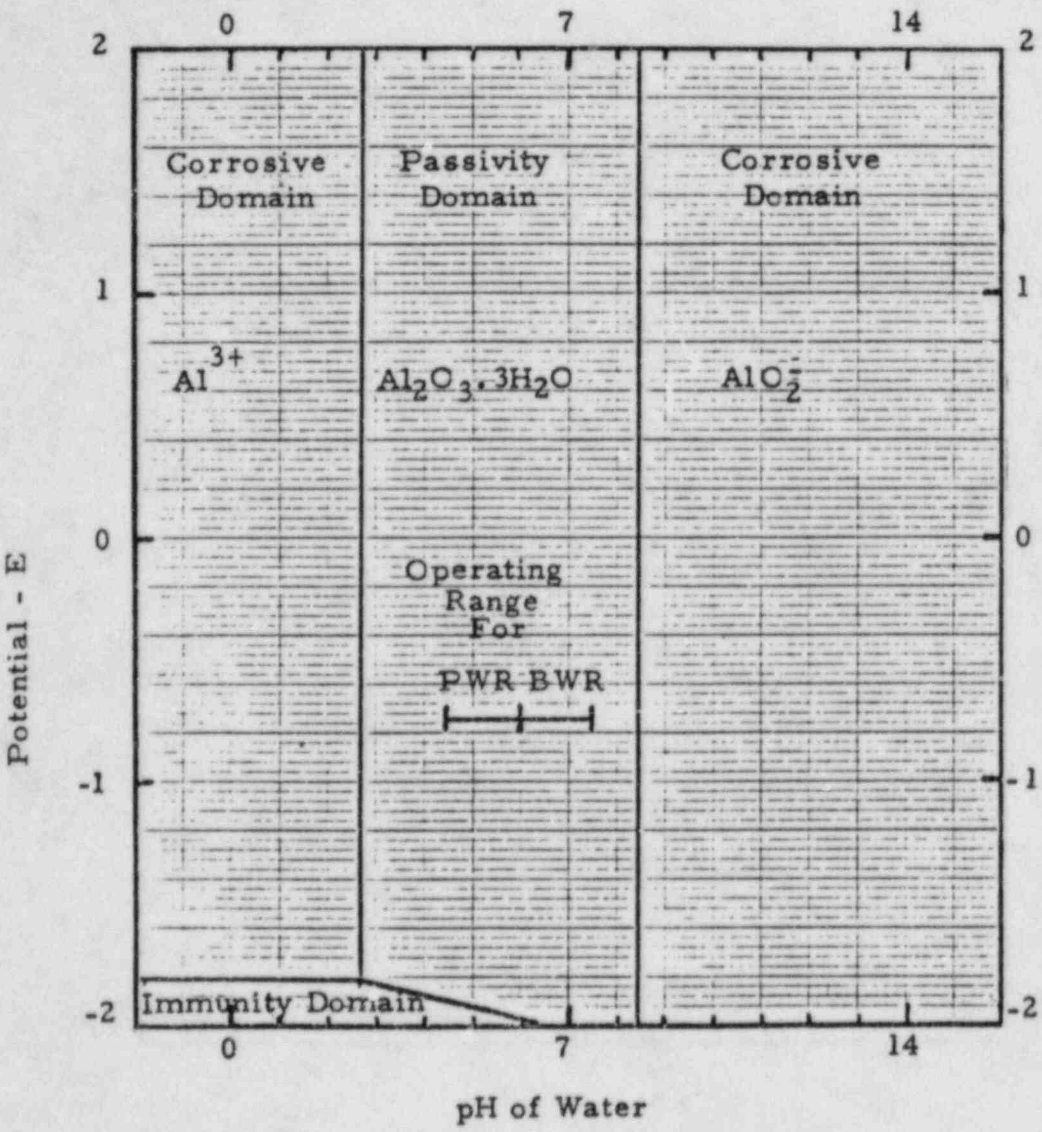
Figure 3

Potential Versus pH Diagram

For Aluminum-Water System

At 60°C (140°F) (5)



Galvanic Corrosion. Galvanic corrosion is associated with the current of a galvanic cell consisting of two dissimilar conductors in an electrolyte. The two dissimilar conductors of most interest in this discussion are aluminum and stainless steel while in an electrolyte similiar to the pool water from either a BWR or PWR. There is less galvanic current flow between the aluminum-stainless steel couple than the potential difference would indicate because of the greater than normal resistance at the metal-liquid interface on stainless steel which is known as polarization.[6] It is because of this polarization characteristic that stainless steel is compatible with aluminum in all but severe marine, or high chloride, environmental conditions. Test data for aluminum coupled with 304 stainless steel in 5.0 pH water at $100^{\circ}C$ ($212^{\circ}F$) with flow rates ranging from 0.5 fpm to 81 fps show weight losses of 0.1 to 0.2 mpy and randomly spread pits that were not of major consequence.[8] This performance indicates a projected service life much greater than forty years.

Pitting Corrosion. Pitting corrosion is the forming of small sharp cavities in a metal surface. The first step in the development of corrosion pits is a local destruction of the protective oxide film. Pitting will not occur on commercially pure aluminum when the water is kept sufficiently pure, even when the aluminum is in electrical contact with stainless steel.[9]

Pitting of aluminum has been observed when in contact with stainless steel where the electrolyte can stagnate and the conductivity of the electroylyte increases.

This pitting has not been significant in spent fuel environments and it is not likely that pitting of the aluminum would have any influence on the neutron shielding performance of the Boral.[4]

Crevice Corrosion. Crevice corrosion is the corrosion of a metal that is caused by the concentration of dissolved salts, metal ions, oxygen or other gases in crevices or pockets remote from the principal fluid stream, with a resultant build up of differential galvanic cells that ultimately cause pitting. Testing has confirmed that after 2000 hours, under a controlled environment, the Boral and 304 stainless steel combination exhibited little or no corrosion of the aluminum cladding of the Boral. In a separate 2000 hour test at 90° to 180°C the maximum pit depth of corrosion of the Boral surface was reported at less than five mils giving a projected life much greater than forty years.[8]

Intergranular Corrosion. Intergranular corrosion is corrosion occurring preferentially at grain boundaries or closely adjacent regions without appreciable attack of the grains or crystals of the metal themselves. Intergranular corrosion does not occur with the commercially pure aluminum (alloy 1100) and other common work hardening alloys.

Stress Corrosion. Stress corrosion cracking is failure of the metal by cracking under the combined action of corrosion and high stresses approaching the yield stress of the metal. The 1100 alloy used in Boral is not susceptable to stress corrosion and Boral is seldom if ever subjected to high stresses when used as a neutron shield in a spent fuel rack.

Corrosion Monitoring System. A corrosion monitoring system is a program whereby a series of surveillance samples are placed in the spent fuel radiation and pool water environment and are periodically examined for physical and chemical changes. It is important the physical configuration of the samples be carefully selected so they are representative to the construction and design of the spent fuel racks and are positioned in the pool to be exposed to representative pool conditions and radiation environment. The physical and chemical characteristics of the samples must be precisely established before insertion into the pool so precise quantative comparisons can be made after each exposure period. The procedure for the manufacture and testing of surveillance samples recommended by Brooks & Perkins is contained in Procedure No. BPS-454. For further discussion on corrosion see Brooks & Perkins Bulletin No. 101.

RADIATION RESISTANCE

Boral has the ability to shield thermal neutrons from nuclear fuel assemblies without physical change or degradation of any sort from the accompanying exposure to heat and gamma radiation. This ability is attributable to the fact that Boral is a thermal neutron shield that contains no organic nor polymeric type binders which undergo extensive crosslinking and oxidative scission type degradations from both heat and radiation exposure. Boral utilizes an all metallic binder which is stable and unchanged under long-term gamma and neutron irradiation and heat up 540°C (1000°F).

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Boral, in addition to having the longest history of use in spent fuel storage applications (since 1965), has been subjected to accelerate irradiation tests which fully support the stability of Boral under these environments. Boral test specimens have been exposed to cumulative doses of 10^{11} rads gamma and 5.3×10^{19} neutrons per sq cm in demineralized and borated water without detectable out-gassing attributable to Boral or any decernible physical changes.

The testing referred to was performed at the Phoenix Memorial Laboratory of the University of Michigan using the Ford Nuclear Reactor. The purpose of the test was to determine changes to physical and chemical properties of Boral as a result of irradiation under conditions similiar to those encountered in PWR and BWR spent fuel storage pools. The data recorded during this testing effort is available upon request and includes the following:

Total radiation exposure and residual radioactivity
Dimensions
Weight
Specific gravity
Hardness
Mechanical strength
Neutron attenuation
Solution boron content, pH, conductivity, and leachable halogens

During irradiation, gas evolution rate, total volume of gas evolved, and gas composition were determined. The Boral samples were irradiated in air, demineralized water, and 2000 ppm borated water which simulate both the vented and sealed enclosure of Boral in both PWR and BWR spent fuel storage environments.

The test results show conclusively there is no out-gassing from Boral when irradiated in dry air. The same was also true for boron carbide powder in a dry aluminum sample container. This clearly show that Boral is unaffected by radiation exposure making Boral a neutron absorber that can be safely exposed while being contained in a sealed enclosure. This characteristic of Boral, no out-gassing from irradiation, also clearly shows that the source of the evolved gases when water was added to the sample containers with Boral has to be from the water itself. There are two mechanism by which water will evolve gases under these circumstances and only one of which requires a radiation environment. The one mechanism requiring a radiation field is the hydrolysis of the water. The disassociation of water into its hydrogen and oxygen elements also requires the presence of free radical scavengers which could well be the boron carbide powder, impurties within the powder, impurties in the water, or surface irregularities on the Boral sample. Gases evolved by hydrolysis would be a hydrogen-oxygen gas mixture in a 2:1 ratio.

The other mechanism by which water will evolve gases is from the chemical reactions between aluminum and water. The sample containers were made of aluminum with an internal surface area of approximately 9.5 sq. in. The surface area of the aluminum cladding on the Boral samples were approximately 3.5 sq. in. The gas released from the water-aluminum reaction is hydrogen as shown in the following reaction:

$$Al^{+++} + H_2O \longrightarrow Al(oH)^{++} + H^+$$

$$2\ Al + 6\ H_2O \longrightarrow Al_2O_3 \cdot 3H_2O + 6H^+ + 6\ \text{electrons}$$

$$2\ H^+ + 2\ \text{electrons} \longrightarrow H_2 \uparrow \qquad (5)$$

The water-aluminum reactions are self-limiting because the surface of the aluminum becomes passive by the formation of a protective and impervious coating making further reaction impossible until that coating is removed by mechanical or chemical means.

The volumes and types of gases collected from the Boral in demineralized and borated water strongly indicate the gases resulted from one or both of the two described mechanisms and did not result from cross linking or oxidative scission of any of the Boral materials.

In summary Boral does not out-gas or change physically or chemically as a result of exposure to gamma radiation. Water in contact with aluminum will release hydrogen chemically until the aluminum surface is passivated and water will disassociate through hydrolysis from gamma radiation. It is therefore necessary to provide a means for venting the hydrogen and oxygen gases if water is allowed to come in contact with Boral in spent fuel storage applications.

NEUTRON SHIELDING PERFORMANCE

The thermal neutron shielding performance of Boral is obtained from the B^{10} isotopes contained within the boron carbide particles in its core. This performance is directly related to the amount of boron carbide provided and the spacial relationship between the particles of boron carbide. Figure 4 shows the actual performance of Boral as compared to an ideal (unobtainable) layer of B^{10} isotopes. The shielding performance is measured as a neutron attenuation factor and is plotted against the surface density of B^{10} isotopes in grams per square centimeter. For further discussion on the shielding properties of Boral see Brooks & Perkins Bulletin No. 100. The neutron shielding performance of Boral was unaffected after exposure to 1.03 x 10" rads gamma and 5.3×10^{19} thermal neutrons per sq cm.

Boron and Halogen Leachability. The boron leachability and the halogen leachability was evaluated for Boral during irradiation testing conducted at the University of Michigan. The test solutions were analized for boron and halogen contents before and after radiation exposure when sufficient solution was remaining after the test. The volume of solution was reduced to zero in some cases by the radiation. The analysis of the test solutions showed no increase in boron or halogen that cannot be accounted for by the decrease in test solution volume or pickup of the soluble boron on the external edges of the Boral. The boron carbide is allowed to contain, by the ASTM Specification C750-80, up to a maximum of three percent (3.0%) soluble boron in the form of boric oxide (B_2O_3).

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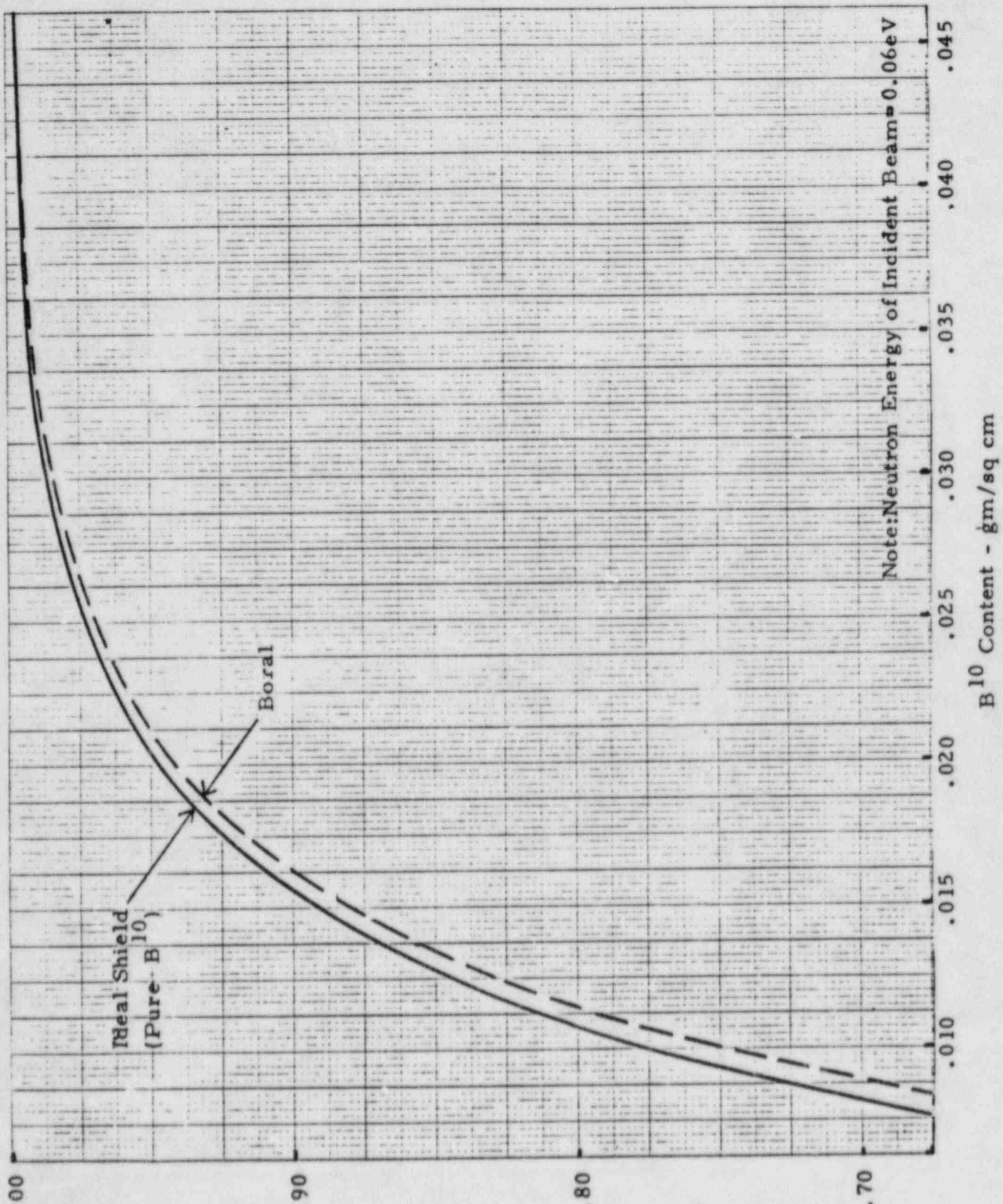
Figure 4

Neutron Attenuation Versus B^{10} Content



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The amount of boron carbide that can come in contact with water is limited to that which is confined to the outer edges of the Boral panel. This wettable amount of boron carbide is of course influenced by the geometrical size and shape of the panel but is less than one percent (1.0%) of the total boron carbide contained therein. In any regard, the total boron content of the panel will remain above the specified minimum content in the event the total soluble boron content were somehow lost through dissolution.

Residual Activity. The residual radioactivity of the Boral was measured following the irradiation testing conducted at the University of Michigan. The activation was limited to trace amounts of impurities contained in the boron carbide and aluminum materials from which Boral is produced. The specific results are available upon request.

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Installations Using Boral

I. Spent Fuel Storage Racks

A. Pressurized Water Reactors

	Reactor	Utility	Water Contact	Service Date
1.	Yankee Rowe	Yankee Atomic Electric Co.	Yes	1964
2.	Maine Yankee	Maine Yankee Atomic Power Co.	No	1977
3.	Cook 1&2	Indiana & Michigan Electric Co.	No	1979
4.	Sequoyah 1&2	Tenn Valley Authority	No	1979
5.	Zion 1&2	Commonwealth Edison Co.	Yes	1980
6.	Salem 1&2	Public Service Electric & Gas Co.	No	1980
7.	Bellefonte 1&2	Tenn Valley Authority	No	1981
8.	Yellowcreek 1&2	Tenn Valley Authority	No	Indef.

B. Boiling Water Reactors

	Reactor	Utility	Water Contact	Service Date
1.	LaCrosse	Dairyland Power Coop.	Yes	1976
2.	Pilgrim 1	Boston Edison Co.	No	1978
3.	Monticello	Northern States Power Co.	Yes	1978
4.	Vermont Yankee	Vermont Yankee Nuclear Power	No	1978
5.	Peach Bottom 2&3	Philadelphia Electric Co.	No	1978
6.	Fitzpatrick	Power Authority of State NY	No	1978
7.	Cooper	Nebraska Public Power District	Yes	1979
8.	Duane Arnold 1	Iowa Electric Light & Power Co.	No	1979
9.	Susquehanna 1&2	Pennsylvania Power & Light Co.	No	1979
10.	Perry 1&2	Cleveland Electric Illuminating Co.	No	1979
11.	Limerick	Philadelphia Electric Co.	No	1980
12.	Browns Ferry 1,2, &3	Tenn Valley Authority	Yes	1980
13.	Dresden 1,2, &3	Commonwealth Edison Co.	Yes	1981
14.	Hatch 1&2	Georgia Power Co.	Yes	1981
15.	Brunswick 1&2	Carolina Power & Light Co.	Yes	1981
16.	Clinton	Illinois Power Co.	Yes	1981
17.	Hartsville 1&2	Tenn Valley Authority	Yes	Indef.
18.	Phipps Bend 1&2	Tenn Valley Authority	Yes	Indef.

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References

1. Yankee Rowe, Rowe, Mass., Boral Spent Fuel Storage Rack in 800 ppm boron max. water, installed Aug. 1964, removed in 1981, small amount of pitting, good structural integrity (2).

2. F.M. Kustas, S.O. Bates, B.E. Opitz, A.B. Johnson Jr., J.M. Perez Jr., R.K. Farnsworth, "Investigation of the Condition of Spent Fuel Pool Components", Battelle-Pacific Northwest Laboratory, PNL-3513/UC-85 Sept. 1981 pg. 5.

3. Brookhaven Medical Research Reactor, Boral in fuel storage area since Jan. 1959, in demineralized water, no loss of boron carbide after more than 19 years (4).

4. C. Czajkowski, J.R. Weeks, and S.R. Protter, "Corrosion of Structural and Poison Material in Spent Fuel Storage Pools", Paper No. 163 presented at Corrosion 81, Apr. 1981, Toronto, Canada.

5. D.D. Mac Donald and P. Butler, "The Thermo-dynamics of the Aluminum - Water System at Elevated Temperatures" Corrosion Science 1973 Vol. 13 pgs. 264, 265 & 266.

6. K.R. Van Horn, "Aluminum", American Society for Metals, 1967 Vol. 1 pgs. 211, 220 & 221.

7. T. Lyman, "Metals Handbook" 8th Edition, 1961 Vol. 1 pg. 930.

8. J. L. English and J.C. Griess, "Dynamic Corrosion Studies for High Flux Isotope Reactor" ORNL-TM-1030 Sept. 1966 pgs. 1, 2, 3, 4, 23, 26, 27 & 31.

9. K. Videm, "Pitting Corrosion of Aluminum in Contact with Stainless Steel" Institute for Atomenergi, Kjeller Research Establishment, Lillestrom, Norway.

10. E. Deltombe, C. Vanleugenhaghe and M. Pourbaix, "Aluminum", pg. 172

DSER Open Item No. 143 (DSER Section 9.1.5)

OVERHEAD HEAVY LOAD HANDLING

We cannot conclude that the overhead heavy load handling systems are in compliance with the Phase I and Phase II criteria contained in NUREG-0612 until the applicant provides an acceptable response to the guidelines. The overhead heavy load handling systems do not meet the acceptance criteria of SRP Section 9.1.5. We will report resolution of this item in a supplement to this SER.

Guideline 1 - Safe Load Paths [NUREG-0612, Article 5.1.1(1)]

(Reference: DSER, Appendix A, Section 2.3.1)

Recommendation a. Submit drawings to show the locations of the cranes in relation to the safe load paths.

RESPONSE

Table 9.1-10 provides the figure number of the plant equipment location drawing and the area on that drawing, defined by column lines, that shows the floor location below each HCGS heavy load handling crane or hoist. For example, the equipment location drawing for the personnel airlock hoist (Table 9.1-10, item 2) is Figure 1.2-28, and the floor location that envelopes the area below the airlock hoist monorail is bounded by column lines P, R, 20R, and 23R. To show the relative elevations between the loads and the floor below the loads, Section 9.1.5.2 has been revised to provide the elevation of the rail(s) and of the hook in its raised position for each non-exempt crane listed in Table 2.1 of HCGS DSER, Appendix A, except the Reactor Water Cleanup Filter/ Demineralizer Hoist, RCIC Pump and Turbine Hoist, Turbine Building Bridge Crane, Turbine Generator Auxiliary Crane, and the Demineralizer Removal Hoist which are discussed below. The location of a non-exempt crane in relation to its safe load path(s) can be determined by referring to the appropriate equipment location drawing and floor area as provided in Table 9.1-10 together with the rail and hook elevations provided in revised Section 9.1.5.2.

The Reactor Water Cleanup F/D Hoist has been reclassified from non-exempt to exempt status because there is no safe shutdown or decay heat removal equipment beneath the load path. This interpretation is based on the guidance provided in Item 2.1.1 of Enclosure 3 to the NRC letter to PSE&G of December 22, 1980 (Reference 3 of HCGS DSER, Appendix A.) The design of this heavy load handling system has been changed to employ two hoists instead of one, a shorter monorail, new load paths, and a removable stop on the monorail. These changes provide the basis for the reclassification. The two new hoists, 1AH220 and 1BH220, replace hoist 10H213. The shortened monorail extends 8 ft-10 in. beyond the south wall of the F/D cells (or 11 in. north of column 15R). Figure 1.2-31 has been revised to show this. The new load paths do not extend beyond the south wall of the F/D cells because the revised plan is to stack the four shield blocks for a given cell onto the roof of the adjacent cell at elevation 178 ft-6 in., instead of lowering them to the elevation 162 ft level as originally planned. The orientation and size of the shield blocks is such that they cannot freely pass through the available opening from elevation 178 ft-6 in. to elevation 162 ft. The available opening is a rectangle with a north-south dimension of 8 ft-10 in. (constrained by the length of the shortened monorail) and a east-west dimension of 7 ft-9 in. (constrained by the dryer-separator pool wall on the west and a floor framing beam at elevation 178 ft6 in. on the east). The two upper blocks must be lifted simultaneously by the two hoists. Together, they form a rectangle with a 11 ft. north-south (compared with 8 ft-10 in. available opening) and a 10 ft-9 in. east-west (compared with 7 ft-9 in. available opening) dimension. The two lower blocks are lifted individually. Each is a rectangle with a 4 ft-3 in. north-south (compared with 8 ft-10 in.) and a 8 ft. 9 in. east-west (compared with 7 ft-9 in.) dimension. The removable stop will prevent a shield block from being carried over the opening. The load handling procedure for the blocks will require the operator to verify that the stop is bolted in place before beginning the lift. The FSAR has been revised (9.1.5.2.2.c, 9.1.5.3.3.c, Table 9.1-10, Table 9.1-12, and Figure 9.1-38) to describe the new design.

The RCIC Pump and Turbine Hoist has been reclassified from non-exempt to exempt status because it does not handle heavy loads during maintenance. The RCIC turbine case weight of 2.35 tons, originally given in Table 9.1-12, was the total weight of the RCIC turbine, including baseplate and stop valve. The maximum weight to be lifted is the casing, which weighs 785 pounds. After the upper half of the turbine case (785 pounds), the next heaviest RCIC turbine maintenance loads are the stop valve (400 pounds) and the rotor (325 pounds). Sections 9.1.5.2.2.3 and 9.1.5.3.3.e, Tables 9.1-10 and 9.1-12, and Figure 9.1-34 have been revised to describe the exempt status of this hoist.

DSER Open Item No. 143 (Cont'd)

The Turbine Building Bridge Crane, Turbine Generator Auxiliary Crane, and Demineralizer Removal Hoist have also been reclassified from non-exempt to exempt status because there is no safe shutdown or decay heat removal equipment beneath their load paths as originally stated in 9.1.5.3.3.k, z, and aa respectively. Tables 9.1-10 and 9.1-12, and Figure 9.1-39 have been revised to be consistent with the FSAR text for these hoists.

Note also that the outboard MSIV Hoist, 10H214, has been incorporated into the design of the new Main Steam Tunnel (MST) underhung crane. The FSAR has been revised (9.1.5.2.2.f, 9.1.5.3.3.f, Table 9.1-10, Table 9.1-12 and Figure 9.1-3) to describe the new design.

Rail elevations, but not raised hook elevations, have been added to Section 9.1.5.2.2 for the CRD Service Hoist (9.1.5.2.2.h), SACS Pumps Hoist (9.1.5.2.2.11), and SACS Heat Exchanger Hoist (9.1.5.2.2.mm) because the hook elevations are not known. The raised hook elevations are not known because instead of dedicated hoists for each service, the hoists will be transferred from other locations as required. The hook heights are expected to be approximately two (CRD), six (SACS Pumps), and four (SACS heat exchangers) feet below the corresponding rail elevations.

Recommendation b. Check the loads carried by the rigging beam hoists for servicing CRD, SACS pump, and SACS heat exchanger to determine the exempt or non-exempt status of these load handling systems, and list the non-exempt systems in Table 2.2.

RESPONSE

The loads for the CRD service hoist are identified in Section 9.1.5.2.2.h as a control rod (450 pounds), neutron monitoring cask (<1150 pounds), and unspecified CRD maintenance equipment (up to 2000 pounds). Control rods and the neutron monitoring cask are not heavy loads for HCGS. A maximum weight of 2000 pounds is chosen for the unspecified CRD maintenance equipment because that will be the capacity of the hoist that will be used on the CRD service monorail. Because a dedicated hoist was not purchased for this service, a hoist will be transferred from another location when one is needed. None of the three

DSER Open Item No. 143 (Cont'd)

exclusion criteria of Table 9.1-10 apply to this hoist. Therefore, it would be classified as non-exempt if it handled a maintenance load equal to or greater than 1200 pounds. This hoist has been added to FSAR Tables 9.1-10 and 9.1-12, instead of to Table 2.1 (not Table 2.2 as typed on page 22) of DSER Appendix A, because the Table 2.1 format is NRC controlled, and is a summary of information first provided in Table 9.1-10.

The loads for the SACS pumps hoists are identified in Section 9.1.5.3.3.11 as the SACS pump motors (1155 pounds). The revised motor weight is 6160 pounds. Section 9.1.5.3.3.1 has been revised to incorporate the new weight. A dedicated hoist was not purchased for this service because the heaviest anticipated SACS pump maintenance load is the upper half of the pump casing (825 pounds). Therefore, this hoist does not routinely handle heavy loads. Instead, a hoist will be borrowed from another location when needed. None of the three exclusion criteria of Table 9.1-10 apply to this hoist. Therefore, it would be classified as non-exempt if it lifts a heavy load. This hoist has been added to Tables 9.1-10 and 9.1-12.

The load for the SACS heat exchanger hoists is identified in 9.1.5.3.3.mm as a SACS heat exchanger return end cover. The return end cover weighs 18,400 pounds. Dedicated hoists were not purchased for this service. Instead, hoists will be borrowed from another location when needed. None of the three exclusion criteria of Table 9.1-10 apply to these hoists. Therefore, they are classified as non-exempt. These hoists have been added to Tables 9.1-10 and 9.1-12.

Recommendation c. Provide the missing information concerning the safe load paths for the non-exempt cranes.

RESPONSE

The missing information has been added to Section 9.1.5. The load paths for the CRD service hoist, the SACS pumps hoist, and the SACS heat exchanger hoists have been added to Figure 9.1-35.

Because the turbine building bridge crane and the turbine-generator auxiliary crane have been reclassified as exempt, (see response to Recommendation b above) their safe load path drawings have not been added to Section 9.1.5 as requested in Item B.2 on Page 9 of DSER, Appendix A. Because the RCIC Pump and Turbine Hoist has been reclassified as exempt, it's load path has been deleted from Figure 9.1-34. Because the Reactor Water Cleanup Filter-Demineralizer Hoist and the Demineralizer Removal Hoist have been reclassified as exempt, their safe load path drawings (Figures 9.1-38 and 9.1-39) have been deleted.

The load paths for the seven loads listed in Item B.3.a of DSER, Appendix A have been added to Figure 9.1-32. The 4-ft by 4-ft. hatch and the 10-ft by 10-ft hatch paths were added to Sheet 1 of Figure 9.1-32; the spent fuel gates path and the flux monitor shipping crate paths were added to Sheet 2; Sheet 6 was added for the refueling bellows guard ring path; Sheet 7 was added for the channel handling boom crane path; and Sheet 8 was added for the RPV head stud rack path.

The load path for the new fuel vault covers has been deleted from Figure 9.1-32, Sheet 1, because the covers are not heavy loads. Each of the three sections is made of 0.25 in thick steel plate and weighs less than 900 pounds.

Safe load paths for three additional polar crane heavy loads that are listed in Table 9.1-12, but not mentioned in Item B.3.a of DSER, Appendix A, have been added to Figure 9.1-32. The main hoist load block, auxiliary hoist load block, and spent fuel cask yoke are shown on new Sheets 9, 10, and 11, respectively.

The dryer-separator sling has been added to Table 9.1-12, and the load path is shown on new Sheet 12. The spent fuel rack modules and the fuel rack lifting fixture have been added to Table 9.1-12, and the load paths are shown on revised Sheet 4 of Figure 9.1-32. The reactor well shield plug sling and the dryer-separator pool plug grapple have been added to Table 9.1-12, and the load paths are shown on new Sheet 9 of Figure 9.1-32.

The justification in Section 9.1.5.6 for the HCGS deviation from Acceptance Criterion 2 of SRP Section 9.1.5, has been revised to clarify the HCGS position that load paths will not be painted on the floor. Instead, the alternative method of using a signalman and temporary load paths as suggested in Item C.4 on Page 11 of DSER, Appendix A, will be used.

DSER Open Item No. 143 (Cont'd)

Guideline 2 - Load Handling Procedures [NUREG-0612, Article 5.1.1(2)]

(Reference: DSER, Appendix A, Section 2.3.2)

Recommendation: Provide evidence to show that the load handling procedures have been developed.

RESPONSE

As agreed in the May 23, 1984 conference call between the applicant and the NRC, the load handling procedures described in FSAR Design Basis Section 9.1.5.1.i will be developed before fuel load. HCGS considers Section 9.1.5.1.i to be a commitment to comply with Article 5.1.1(2) of NUREG-0612.

Guideline 3 - Crane Operator Training [NUREG-0612, Article 5.1.1(3)]

(Reference: DSER, Appendix A, Section 2.3.3)

Recommendation: Provide information on the status or plan for crane operator training in accordance with Chapter 2-3 of ANSI B30.2-1976.

RESPONSE

As discussed in the May 23, 1984 conference call between the applicant and the NRC, crane operators will be trained, qualified, and conduct themselves in accordance with Chapter 2-3 of ANSI B30.2-1976, as stated in FSAR Design Basis Section 9.1.5.1.j. HCGS considers Section 9.1.5.1.j to be a commitment to comply with Article 5.1.1(3) of NUREG-0612.

DSER Open Item No. 143 (Cont'd)

RESPONSE

Guideline 4 - Special Lifting Devices [NUREG-0612, Article 5.1.1 (4)]

(Reference: DSER, Appendix A, Section 2.3.4)

Recommendation a. Recalculate the stress design factors based on the combined maximum dynamic and static load. These factors must be equal or greater than those specified in ANSI N14.6-1976.

RESPONSE

The recalculated stress design factors based on the combined maximum dynamic and static loads are provided in revised Table 9.1-14.

The title of Table 9.1-14 has been changed from "Hope Creek Polar Crane Special Lifting Devices and Slings" to "Hope Creek Special Lifting Device Factors of Safety". The fuel pool slot plug sling (Item 7) has been deleted from the table to reflect design evolution. The slot plugs will be lifted with conventional slings instead of with a dedicated lifting device. Two new special lifting devices (SLD), the personnel air lock strongback (Item 9) and the fuel rack lifting fixture (Item 10) have been added to the table. Item 9 was added when the scope of Table 9.1-14 was expanded to include all HCGS SLD instead of just the polar crane SLD. Item 10 is supplied with the spent fuel racks. The HCGS plan is to install partial spent fuel storage capacity during construction and additional racks as needed after plant operation begins. Item 10 would be used to install these spent fuel rack modules.

Sections 9.1.5.3.2 and 9.1.5.3.3.a have been revised to be consistent with the recalculated factors of safety provided in revised Table 9.1-14. The reference factors of safety are obtained from ANSI N14.6-1978, instead of N14.6-1976 as stated in Item 4.a on Page 25 of DSER, Appendix A, because Paragraph 5.1.1(4) of NUREG-0612 invokes the 1978 version.

DSER Open Item No. 143 (Cont'd)

The missing safety evaluation information for the RPV stud tensioner sling, noted in Item B on Page 14 of DSER, Appendix A, is provided in revised Section 9.1.5.3.2.

As stated in Section 9.1.5.3.2, a single-failure proof conventional sling is used to lift the fuel pool gates. Because it is a conventional sling and not a special lifting device, it does not appear in Table 9.1-14.

Recommendation b. Address the requirements of ANSI N14.6-1976 in addition to the requirements for stress design factors.

RESPONSE

As shown in revised Table 9.1-14, the stress design factors for all but two of the Hope Creek special lifting devices meet or exceed the values of 3 versus yield strength and 5 versus ultimate strength required by ANSI N14.6-1978. The design of the two Hope Creek lifting devices (dryer-separator sling and RPV service platform sling) that do not meet the safety factor criteria is the same as the design of the corresponding Washington Nuclear Plant No. 2 and Limerick Generating Station special lifting devices. Therefore, as discussed in the August 24, 1984, telephone call between the applicant and the NRC, no additional information regarding compliance with ANSI N14.6-1978 is provided.

DSER Open Item No. 143 (Cont'd)

Guideline 5 - Lifting Devices (not specially designed) [NUREG-0612, Article 5.1.1(5)]

(Reference: DSER, Appendix A, Section 2.3.5)

Recommendation: Provide information concerning the installation and use of the lifting devices as required.

RESPONSE

The heavy load handling system Design Bases have been revised to provide the requested information in FSAR, Section 9.1.5.1.n. HCGS considers Section 9.1.5.1.n to be a commitment to comply with the provisions of Article 5.1.1(5) of NUREG-0612.
As discussed in the September 6, 1984 telecon between the applicant and the NRC, a dynamic load will not be added to the static load when a sling is selected for use with a hoist that has a maximum hoisting speed equal to or less than 30 ft./min.

DSER Open Item No. 143 (Cont'd)

Guideline 6 - Crane Inspection, Testing and Maintenance [NUREG-0612, Article 5.1.1(6)]

(Reference: DSER, Appendix A, Section 2.3.6)

Recommendation: Provide information for inspecting, testing, and maintaining the cranes including those that are not listed in FSAR, Table 9.1-10, but may carry heavy loads over safety related equipment.

RESPONSE

All cranes and hoists at HCGS that may carry heavy loads over safety related equipment, including the CRD service hoist, SACS pumps hoist and SACS heat exchanger hoists which were not originally listed, are now listed in revised Table 9.1-10. The procedure for inspecting, testing, and maintaining each of these non-exempt cranes (those not identified by exclusion criteria A, B, or C) will comply with Chapter 2-2 of ANSI B30.2-1976. HCGS considers FSAR Section 9.1.5.1.k., together with revised Sections 9.1.5.4.1.3 and 9.1.5.4.2.3, to be a commitment to comply with Article 5.1.1(6) of NUREG-0612.

DSER Open Item No. 143 (Cont'd)

Guideline 7 - Crane Design [NUREG-0612, Article 5.1.1(7)

(Reference: DSER, Appendix A, Section 2.3.7)

Recommendation: Provide information explicitly directed at complying with the guidelines of ANSI B30.2-1976, Chapter 2-1 and of CMAA-70.

RESPONSE

As agreed in the May 23, 1984 conference call between the applicant and the NRC, the requested information is already provided in Table 9.1-10. The Design Standard column identifies the standards that were applied for the design of each crane and hoist in revised Table 9.1-10.

The HMI 100 design standard was added for Items 3,18,19,31 and 37, and Note 4 (use of ANSI B30.17) was added for Items 31 and 32 to correct their inadvertent omission from the original submittal. The HMI 100 standard was deleted for Item 4 (reactor water cleanup F/D hoist) because the motorized hoists were replaced with manual hoists. The ANSI B30.2 and B30.17 design standards were added to Item 7 because the outboard MSIV hoist was incorporated in the expanded main steam tunnel underhung crane design.

Section 3.3 Interim Protection [NUREG-0612, Article 5.3]

(Reference: DSER, Appendix A, Sections 2.4.2 and 2.4.3)

Recommendation: The Interim Protection Measures 2 through 6 should be initiated to provide safe operation of the cranes before implementation of the guidelines of NUREG-0612, Article 5.1 is completed.

RESPONSE

Interim Protection Measures 2 through 6 are not applicable to HCGS because the plant is not operational. HCGS's intent is that implementation of the guidelines of NUREG-0612, Article 5.1 will be completed before the plant is operational.

HCGS's detailed position with respect to Measures 2 through 6 is clarified below.

Interim Protection Measure 2 (load paths)

Safe load paths have been defined per the guidelines of Section 5.1.1(1) of NUREG-0612, as described above in the response to DSER, Appendix A, Section 2.3.1, and in the associated revision of FSAR Section 9.1.5.6. Therefore, this measure will not be required when the plant becomes operational.

Interim Protection Measure 3 (procedures)

Procedures will be developed and implemented per the guidelines of Section 5.1.1(2) of NUREG-0612 before fuel load, as described above in the response to DSER, Appendix A, Section 2.3.2. Therefore, this measure will not be required when the plant becomes operational.

Interim Protection Measure 4 (operators)

Operators will be trained, qualified, and will conduct themselves per the guidelines of Section 5.1.1(3) of NUREG-0612, as described above in the response to DSER, Appendix A, Section 2.3.3. Therefore, this measure will not be required when the plant becomes operational.

Interim Protection Measure 5 (crane maintenance)

Cranes will be inspected, tested, and maintained in accordance with the guidelines of Section 5.1.1(6) of NUREG-0612, as described above in the response to DSER, Appendix A, Section 2.3.6, and in the associated revisions to FSAR Sections 9.1.5.4.1.3 and 9.1.5.4.2.3. Therefore, this measure will not be required when the plant becomes operational.

DSER Open Item No. 143 (Cont'd)

Interim Protection Measure 6 (heavy loads over core)

As described above in the response to DSER, Appendix A, Section 2.3.2, specific load handling procedures for each load will be developed before fuel load. During procedure development, rigging or lifting device installation and load movement requirements will be reviewed to assure that sufficient detail and clear, concise instructions are provided. As stated in revised FSAR Section 9.1.5.6, the polar crane signalman will review the specific load handling procedure before each lift. This will supplement the original procedure development review for detail, clarity, and conciseness.

As stated above in the response to DSER, Appendix A, Section 2.3.6, cranes will be inspected in accordance with Chapter 2-2 of ANSI B30.2-1976. Chapter 2-2 requires periodic visual inspection of the crane load bearing components. As stated above in the response to DSER, Appendix A, Section 2.3.4, Recommendation b, the special lifting devices comply with ANSI N14.6-1978 as described in Table 9.1-19. That compliance includes periodic visual inspection of the load bearing welds and critical areas in accordance with Paragraphs 5.3.1(2), 5.3.6, and 5.3.7 of ANSI N14.6-1978. As stated above in the response to DSER, Appendix A, Section 2.3.5, slings will be used in accordance with the guidelines of ANSI B30.9-1971. Such use includes periodic visual inspection of slings in accordance with Sections 9.1-8 (steel chain), 9.2-8 (wire rope), 9.3-7 (metal mesh) 9.4-6 (fiber rope), and 9.5-6 (synthetic webbing) to identify flaws or deficiencies that could lead to failure.

As stated above in the response to DSER, Appendix A, Section 2.3.6, cranes will be maintained in accordance with Chapter 2-2 of ANSI B30.2-1976. Chapter 2-2 requires repair and replacement of defective components. As stated above in the response to DSER, Appendix A, Section 2.3.4, Recommendation b, the special lifting devices comply with ANSI N14.6-1978 as described in Table 9.1-19. That compliance includes repair and replacement of defective components in accordance with Paragraphs 5.4.1 and 5.4.2 of ANSI N14.6-1978. As stated above in the response to DSER, Appendix A, Section 2.3.5, slings will be used in accordance with ANSI B30.9-1971. Such use includes repair and replacement of defective components in accordance with Section 9.1-6 (steel chain), 9.2-8(wire rope), 9.3-7 (metal mesh), 9.4-6 (fiber rope), 9.5-6 and 9.5-7 (synthetic webbing).

DSER Open Item No. 143 (Cont'd)

As stated above in the response to DSER, Appendix A, Section 2.3.3, crane operators will be trained, qualified, and will conduct themselves in accordance with Chapter 2-3 of ANSI B30.2-1976. The examination required of each operator for compliance with Chapter 2-3 will include a requirement to demonstrate familiarity with the hand signals specified in Figure 5 of B30.2. The specific load handling procedure for each heavy load handled over the core will require that the crane operator certify his familiarity with the content of that procedure by signing and dating the procedure before the lift.

Therefore, Interim Protection Measure 6 will not be required when the plant becomes operational.

1. ☐ PSAR ☒ FSAR CHANGE NOTICE

3. NO. 572

2. DISCIPLINE Mechanical

4. ORIGINATOR D.R. Geiger

5. DATE 8/23/84

6. REFERENCED SECTIONS OF SAR

9.1.5 Overhead Heavy Load Handling Systems

7. DESCRIPTION OF CHANGE

Revise, add and/or delete the text, tables and figures of Section 9.1.5 as necessary to be consistent with the response to the NRC comments in DSER Appendix A. Also, incorporate several editorial changes.

ATTACHMENTS:

Revised text pages				Revised tables	Revised figures
9.1-70	9.1-85	9.1-94	9.1-109	T9.1-10 (3 pages)	F 9.1-26
-71	-86	-95	-111	T9.1-12 (8 pages)	F 9.1-32 (sheets 1 thru 12)
-77	-87	-96	-113	T9.1-13 (p. 1,3,4,5)	F9.1-33 through 36
-78	-88	-97		T9.1-14	F9.1-38 and 39
-79	-89	-98			F1.2-31
-81	-90	-105			
-84	-91	-106			
	-93				

8. REFERENCED SPECIFICATIONS OR DRAWINGS

None

9. JUSTIFICATION ☒ BELOW ☐ ATTACHED

Required by NRC, i.e. the changes respond to DSER Item 143.a.

10. DISTRIBUTION/INTERFACING DISCIPLINE REVIEW

ORIGINATING GROUP SUPERVISOR
(PQAE FOR CHAPT. 17)
SAR COORDINATOR
PROJECT ENGINEER
CHIEF NUCLEAR ENGINEER (INFO ONLY)

INTERFACING DISCIPLINES:

Civil S.J. / SB for GLS

15. CONCURRENCE (NSSS) DATE

11. APPROVED BY (GROUP SUPV) DATE

12. APPROVED BY SAR COORDINATOR DATE 9/4/84
H.T. Johnston for P. Coico

13. APPROVED BY (PROJECT ENG.) DATE

14. CONCURRENCE (CLIENT) DATE

FORM SAR-1

SFP-20238

ED-16 REV. 2 (8-76)

9.1.5 OVERHEAD HEAVY LOAD HANDLING SYSTEMS

9.1.5.1 Design Bases

a. The overhead heavy load handling systems (OHLHS) are designed to move heavy loads from one location to another within the various plant structures.

b. The OHLHS are designed to safely handle all plant heavy loads that range in weight from a maximum of 150 tons in the reactor building and 220 tons in the turbine building to a minimum of 1200 pounds.

c. The reactor building polar crane main hoist and auxiliary hoist are designed to be single-failure proof in conformance with NUREG-0554 and NUREG-0612.

d. The OHLHS in the reactor building are designed so that releases of radioactive material that could result from damage to spent fuel, due to a postulated heavy load drop, will produce doses that are within 10 CFR 100 limits.

e. The OHLHS in the reactor building are designed so that damage to fuel and fuel storage racks due to a postulated heavy load drop will not result in a fuel configuration that causes K_{eff} to exceed 0.95.

f. The OHLHS in the reactor building are designed so that damage to the reactor vessel or spent fuel pool, resulting from a postulated heavy load drop, will not cause water loss that could uncover spent fuel.

g. The OHLHS are designed so that damage to equipment resulting from a heavy load drop will not prevent safe shutdown of the reactor.

h. The OHLHS are designed to minimize the potential for heavy load drops on spent fuel or safe shutdown equipment by carrying their loads over safe load paths to the extent practical. They are defined in written load handling procedures, and are shown on safe load path drawings.

i. The reactor building polar crane and other OHLHS that handle loads over safe shutdown equipment are operated in compliance with written procedures that include identification of the required equipment, inspections and acceptance criteria required before load movement, sequence of steps to be followed for load movement, definition of safe load path, and any special precautions, for each known load.

j. The OHLHS cranes are operated by operators who are trained, qualified, and conduct themselves in compliance with Chapter 2-3 of ANSI B30.2-1976.

k. The OHLHS cranes are inspected, tested, and maintained in compliance with Chapter 2-2 of ANSI B30.2-1976.

l. The OHLHS crane designs include electrical interlocks and/or mechanical stops to restrict crane travel to those areas that are necessary.

m. The OHLHS cranes are designed to meet the applicable criteria of CMAA-70, and Chapter 2-1 of ANSI B30.2-1976.

Replace with Insert 1

n. Lifting devices are designed to meet the applicable criteria of ANSI B30.9-1971.

o. Special lifting devices are designed to meet the applicable criteria of ANSI N14.6-1978.

9.1.5.2 System Description

The cranes and lifting devices that comprise the OHLHS are described in the following sections. Table 9.1-10 includes a summary of the design data, seismic category, and code or standard used for design and manufacture of each OHLHS crane. It also includes monorails ~~and lifting beams~~ for which no dedicated hoists exist but which are used occasionally for equipment

(Ref. 9.1.5.1.n) INSERT 1

Lifting devices that are not specially designed are installed and used in accordance with the guidelines of ANSI B30.9-1971. For all hoists that have a maximum hoisting speed equal to or less than 30 ft./min., the load used to select the proper sling is the static load. For hoists that have a maximum hoisting speed greater than 30 ft./min., the load used is the sum of the static and maximum dynamic loads. The maximum dynamic load is determined by multiplying the maximum hoisting speed by ½ of 1% of the static load.

The rating identified on the sling is for the static load that corresponds to the maximum total static plus dynamic load. If a sling is restricted to use with only certain hoists, the identification of the acceptable hoists is clearly marked on the sling.

9.1.5.2.1.1 Structural Components

All the structural components and machinery of the reactor building polar crane are designed for a full capacity of 150 tons, with a minimum safety factor of 10 against ultimate failure for the load-carrying parts, including hoist ropes, and the machinery. The calculated stresses of all load-carrying parts are in accordance with the requirements of Crane Manufacturer's Association of America (CMAA) Specification 70.

Structural design of the crane complies with the following seismic loading combinations and criteria:

a. Dead load plus live load plus operating basis earthquake (OBE) resultant stresses are less than the normal AISC code allowable stresses.

b. Dead load plus live load plus safe shutdown earthquake (SSE) resultant stresses are less than 1.5 times the normal AISC code allowable stresses, less than 0.9 yield in bending, 0.85 in axial tension or compression, and 0.5 yield in shear.

The earthquake motion considered consists of two horizontal and one vertical component. The total structural response is predicted by combining the applicable maximum codirectional responses, calculated from the three (two horizontal and one vertical) analyses, using the square root of the sum of the squares (SQRSS) method.

The structural members of the reactor building polar crane are designed for a fatigue loading of 100,000 to 500,000 cycles, with each completed lift representing one cycle. The rotating machinery is designed for a fatigue life expectancy of 2,000,000 cycles, with each rotating component cycle represented by one revolution. Any load below 50% of the crane rated capacity does not reduce the life expectancy of the crane.

9.1.5.2.1.2 Mechanical Components

The crane is of a double-trolley, indoor, electric overhead, bridge crane design. The main trolley layout is shown on maintenance. Table 9.1-11 includes a more detailed listing of the design parameters for just the reactor building polar crane.

Cranes are included in the OHLHS if their capacity is greater than 1200 pounds. This is the designated weight of a heavy load for Hope Creek Generating Station (HCGS). It is defined as the weight of one spent fuel assembly and its handling tool. For HCGS, the 1200-pound value consists of assumed weights for a fuel assembly (650 pounds), a fuel assembly channel (100 pounds), and the refueling platform grapple (450 pounds).

Section 9.1.4 includes a description of those aspects of new fuel receipt and storage, reactor refueling operations, and spent fuel shipment that involve the reactor building polar crane.

9.1.5.2.1 Reactor Building Polar Crane

at elevation 240 feet.

The reactor building polar crane is a bridge crane mounted on a circular rail that is supported by the reactor building superstructure. The bridge consists of two welded box girders with full depth diaphragms. The bridge girders are held together by structural end tie girders. Two dual-wheeled trucks that travel on top of the runway rail support each of the two end tie girders and drive the bridge. The crane is shown on Figure 9.1-21.

Two electric-motor-driven trolleys, one for the main hoist and one for the auxiliary hoist, provide the structural frame support for the polar crane hoisting machinery. The trolleys travel on a single set of rails secured to the tops of the two bridge girders. The main hoist design capacity is 150 tons, and the auxiliary hoist design capacity is 10 tons. Both hoists are single-failure proof. The electric-motor-driven hoists raise and lower their loads using wire rope that is dual-reeved through upper and lower sheaves. The lower sheaves are an integral part of the load block. Each hoist includes a hook that is attached to the load block. The hooks are at elevation 236 feet when they are fully raised.

The design parameters for the reactor building polar crane are listed in Table 9.1-11.

The reactor building polar crane design includes the features described below.

End of travel limit switches on the auxiliary trolley, and at both ends of the bridge, permit it to travel nearly the full length of the bridge, as shown on Figure 9.1-31. Bumpers at each end of the auxiliary trolley travel path, designed for 100% unloaded impact, back up the auxiliary trolley limit switches.

The bridge and trolley limit switches cut the power to their respective drive motors and set the corresponding brakes when they trip.

9.1.5.2.1.6 Thermal Overload Protection

Thermal overload protection is provided for motors on the crane to prevent continuation of motor-stalling torque. In addition, thermal overload warning lights in the operator's cab indicate bridge, trolley, or hoist motor high temperature.

9.1.5.2.2 Other OHLHS Cranes

All plant OHLHS cranes, except the reactor building polar crane, are described below. ~~They are listed in the same order as they appear in Table 9.1-10.~~ The equipment tag numbers are shown in parentheses.

a. Personnel air lock hoist (10H217)

This 30-ton capacity monorail hoist is located above elevation 102 feet in the reactor building. The top of the rail is at elevation 126 feet 4.75 inches, and the hoist hook is at elevation 118 feet 3.25 inches when it is fully raised. It is used to remove nine shield blocks and the drywell personnel air lock (30 tons) during plant shutdown. The upper shield block includes retractable wheels that permit this hoist to tow it forward along the monorail and position it to be lowered by two adjacent 15-ton hoists (item ii.). The 30-ton hoist lifts each of the eight lower shield blocks, moves it a short distance, and lowers it onto a cart. The cart carries the nine blocks out of the reactor building. The personnel air lock strongback, together with the 30-ton hoist, is used when the air lock is moved. When the strongback is not in use, it remains attached to the air lock. The personnel air lock is moved along the monorail and set down on a predetermined spot. A portion of the primary containment suppression pool is located below the load path of this hoist on the next lower elevation.

For each hoist the top of the rail is at elevation 120 ft. 1 in., and the hook is at elevation 116 ft. when it is fully raised.

b. Reactor recirculation pump motor hoist (1AH201, 1BH201)

These 24-ton capacity monorail hoists are located above elevation 102 feet inside the drywell. ~~They~~ Each hoist lifts ~~the~~ a recirculation pump motor (24 tons) out of the pump housing for inspection and repair. To remove the motor, the hoist raises it (in place) about 3 feet to clear the pump. The removal cart is then moved beneath the motor. When the cart is in place, the motor is lowered to rest on the cart and tilted to a horizontal position. It is tilted by simultaneously moving the chain-operated trolley and lowering the lift point. With the motor secured, the cart is pulled out of the containment.

c. Reactor water cleanup filter-demineralizer hoists (~~10H213)~~ 1AH220, 1BH220)

These 10-ton capacity monorail hoists ~~is~~ are located above elevation 178 feet in the reactor building. ~~It is~~ The hoists are used for removal of the four concrete shield blocks above each RWCS filter-demineralizer cell, and the filter tube bundle. The two heaviest blocks weigh 8 tons each. The tube bundle weighs less than 500 pounds and therefore is not a heavy load.

d. HPCI pump and turbine hoists (1AH211, 1BH211)

For each hoist the top of the rail is at elevation 71 ft. 3½ in., and the hook is at elevation 66 ft. 10½ in. when it is fully raised.

These 4-ton capacity monorail hoists are located above elevation 54 feet in the reactor building. They are used during maintenance of the HPCI pump and turbine. There is no lower floor elevation.

e. RCIC pump and turbine hoist (10H212)

This 3-ton capacity monorail hoist is located above elevation 54 feet in the reactor building. It is used during maintenance of the RCIC pump and turbine. There is no lower floor elevation. The heaviest maintenance load (upper half of the turbine case) weighs 785 pounds. Therefore this hoist does not handle heavy loads.

f. Main Steam Tunnel Underhung Crane ~~Outboard main steam isolation valve (MSIV) hoist~~ (10H214, 10H223)

INSERT 2

This crane ~~2 1/2-ton capacity hoist~~ is located above elevation 102 feet in the reactor building. It is used to lift the operator off of each of the four the outboard MSIV's, the four main steam stop valves, and the two motor operated feedwater stop check valves for maintenance.

g. Inboard MSIV hoist (10H219)

INSERT 3

This 2-ton capacity hoist is located above elevation 102 feet in the reactor building. It is used to lift the operator off of the inboard MSIV for maintenance.

h. CRD service ~~rigging beam~~ hoist ~~(future)~~

This monorail ~~rigging beam~~ is located above elevation 102 feet in the CRD maintenance area of the reactor building. The top of the rail is at elevation 116 ft 6 in. It is designed to accommodate a hoist for lifting control rods (450 pounds), CRD maintenance equipment (up to 2000 pounds), and the neutron monitoring cask (less than 1150 pounds). Because a dedicated CRD service hoist was not purchased, one will be borrowed from another location when needed.

i. Vacuum breaker valve removal hoist (10H207)

This 2-ton capacity circular monorail hoist is located above elevation 54 feet in the reactor building. The monorail is located inside the suppression pool chamber for maintenance and removal of the vacuum breaker valves. The valves weigh 912 pounds each and thus do not constitute a heavy load.

j. Main steam line relief valve removal hoist (10H202)

This 1-ton capacity circular monorail hoist is located above elevation 121 feet inside the drywell. The monorail is actually at elevation 135 feet. It is used to remove the main steam line relief valves as required for maintenance. The main steam line relief valves weigh 1100 pounds each and do not constitute a heavy load.

k. Turbine building bridge crane (10H102)

This crane consists of a 220-ton capacity main hoist and a 45-ton auxiliary hoist. It is located above elevation 137 feet in the turbine building. It is used to lift the parts of the turbine-generator. A second crane, identical to 10H102, and originally intended for use with the Unit 2 turbine, travels along the same rails as 10H102. A stator lift beam, supplied with the cranes and designed to be simultaneously supported by the main hoist of each crane, is used to lift the 366-ton stator of the turbine-generator unit.

l. Feedwater heater removal hoist (1AH103, 1BH103)

These portable 24-ton capacity, manually (chain) operated hoists are designed to operate in tandem on one of the nine I-beam monorails located above elevation 120 feet in the turbine building. The beams serve the nine condenser-mounted feedwater heaters. The hoists are used during feedwater heater tube removal.

m. Heating and ventilating equipment removal hoist (10H104)

This 15-ton capacity monorail hoist is located above elevation 171 feet in the turbine building. It is used for moving heating and ventilation equipment through the equipment removal hatch at elevation 137 feet.

n. Motor-generator set hoist (0AH105, 0BH105)

These 15-ton capacity monorail hoists are located above elevation 137 feet in the turbine building. They service and replace components of the two reactor recirculation pump motor-generator sets.

o. Secondary condensate pump hoist (10H106)

This 15-ton capacity monorail hoist is located above elevation 54 feet in the turbine building. It services

(Ref. 9.1.5.2.2.f) INSERT 2

Two parallel manually driven bridge beams connected by end trucks travel on two fixed girders located in the main steam tunnel. The top of the bridge beams is at elevation 140 feet. A manually operated 2.5 ton capacity trolley and hoist (10H214) is mounted on one bridge beam, and a manually operated 3 ton capacity trolley and hoist (10H223) is mounted on the other. The hook is at elevation 137 ft.-4 in. for 10H214, and 137 ft.-3.5 in. for 10H223 when fully raised.

(Ref. 9.1.5.2.2.g) INSERT 3

It is moved between any of the five monorail beams as needed. The top of the rails are at elevation 119 ft.-7.5 in., and the hoist hook is at elevation 117 ft.-5.5 in. when it is fully raised.

the three secondary condensate pumps and their electric motor drivers from one common ~~rigging beam~~ monorail.

p. Reactor feed pump hoist (1AH107, 1BH107, 1CH107)

These 15-ton capacity chain-operated monorail hoists are located above elevation 137 feet in the turbine building. They service the reactor feed pumps and their turbine drivers.

q. Water box removal hoist (10H109, 10H110)

These 12-ton capacity monorail hoists are located above elevation 77 feet in the turbine building. They are used for removal of the condenser water boxes that have inlet and outlet nozzles.

r. Steam packing exhauster hoist (10H115)

This 10-ton capacity chain-operated monorail hoist is located above elevation 77 feet in the turbine building. It is used during removal of the tube bundle from the steam packing exhauster condenser.

s. Steam jet air ejector hoist (1AH117, 1BH117, 1CH117, 1DH117)

These 8-ton capacity chain-operated monorail hoists are located above elevation 77 feet in the turbine building. They are used during removal of tube bundles from the steam air ejector interim and aftercondenser.

t. Water box removal hoist (10H111, 10H112)

These 8-ton capacity chain-operated monorail hoists are located above elevation 77 feet in the turbine building. They are used for removal of the condenser water boxes that do not have inlet and outlet nozzles.

u. Chiller tube removal hoist (10H118)

This 5-ton capacity chain-operated monorail hoist is located above elevation 171 feet in the turbine building. It is used for removal of chiller tube bundles.

v. Emergency air compressor hoist (10H114)

This 4-ton capacity chain-operated monorail hoist is located above elevation 123 feet in the turbine building. It is used to service the emergency instrument air compressor.

w. Main air compressor hoist (00H113, 10H113)

These 3-ton capacity, chain-operated monorail hoists are located above elevation 123 feet in the turbine building. They are used during replacement of the station air compressors and their motor drivers.

x. Vacuum pump water cooler hoist (10H116)

This 2-ton capacity, chain-operated monorail hoist is located above elevation 77 feet in the turbine building. They are used for removal of tube bundles from the mechanical vacuum pump seal water coolers.

y. Heating and cooling coil removal hoist (1AH119, 1BH119)

These 1.5-ton capacity monorail hoist are located above elevation 171 feet in the turbine building. They are used for removal of the cooling and heating coils that are located inside the air supply plenum.

z. Turbine-generator auxiliary crane (00H100)

This 10-ton capacity bridge crane serves the turbine-generator. A set of rails is provided over the turbine-generator above elevation 137 feet. The radwaste area. They lift the WER pump (3300 pounds) and motor (3835 pounds), separately, for maintenance.

ff. Waste evaporator hoist (00H312, 00H313)

These 1-ton capacity, hand-operated monorail hoists are located above elevation 87 feet of the service and radwaste area. They lift the tops of the evaporators and miscellaneous parts.

gg. Diesel generator underhung crane (1AH400, 1BH400, 1CH400, 1DH400) The top of the bridge is at elevation 124 ft. 9 in., and the hoist hook elevation is 122 ft. when it is fully raised.

These 2-ton capacity underhung bridge cranes are located above elevation 102 feet of the control and diesel generator area. They are used to lift and move miscellaneous diesel generator parts and equipment. The four cranes share a single interchangeable hoist.

hh. Intake structure gantry crane (00H500)

The top of the gantry girder rail is at elevation 167 ft 5.5 in. As shown on Figure 9.1-37, the elevation of the 30 ton hook is 16 ft. and of the 15 ton hook is 160 ft. when fully raised.

This gantry crane is located above elevation 123 feet of the intake structure. It has a 30-ton capacity main hoist and a 15-ton capacity auxiliary hoist. The crane's heaviest lifts are parts of the traveling screens (19 tons).

ii. Reactor building personnel lock shield removal hoist (1AH218, 1BH218)

These 15-ton capacity monorail hoists are located above elevation 102 feet in the reactor building. One is located on each side of the personnel air lock hoist (item a.). After the personnel air lock hoist tows it into position, they work in tandem to lower the upper shield block onto a cart that carries it out of the building.

auxiliary crane is moved by the turbine building bridge crane (item k.), as required.

aa. Demineralizer removal hoist (00H302)

This 10-ton capacity monorail hoist is located above elevation 102 feet in the service and radwaste area. It lifts the fuel pool filter demineralizer and liquid radwaste filter elements out of their vessels for maintenance and replacement. The heaviest load (1800 pounds) is a liquid radwaste filter bundle.

bb. Decontamination evaporator hoist (00H305)

This 7-1/2 ton capacity monorail hoist is located above elevation 54 feet of the service and radwaste area. It is used during maintenance of the decontamination solution evaporator. The hoist lifts the top (1.2 tons) and middle (2.9 tons) sections separately and sets them down on predetermined spots.

cc. Equipment decontamination room hoist (00H314)

This 5-ton capacity bridge-type crane is located above elevation 102 feet of the service and radwaste area. It is used for lifting and moving miscellaneous equipment.

dd. Machine shop underhung crane (0AH301, 0BH301, 0CH301, 0DH301)

These 5-ton capacity monorail cranes are located above elevation 102 feet of the service and radwaste area. They are used for lifting and moving miscellaneous plant equipment and parts.

ee. Waste evaporator recirculation (WER) pump hoist (00H309, 00H310)

These 2-ton capacity, hand-operated monorail hoists are located above elevation 54 feet of the service and jj. Solid radwaste monorail hoist (00H316)

This 1-1/2-ton hoist is located above elevation 102 feet in the auxiliary building. It transfers filled 55-gallon radwaste drums from the two extruder/evaporator turntables to the capper/scanner infeed conveyor and replaces them with empty drums.

kk. Solid radwaste bridge crane (00H317)

This 7-1/2-ton double girder crane is located above elevation 102 feet in the auxiliary building. It also serves the radwaste drum storage area loft at elevation 126.5 feet. It moves filled 55-gallon drums within the storage area, unloads the outfeed conveyor, assists in removing the shipping cask lid, and in truck loading.

ll. SACS pumps ~~rigging beam~~ hoist ~~(future)~~

Fifteen-ton capacity monorails ~~rigging beams~~ above the SACS pumps are designed to accommodate hoists for removal of the pump motors. One monorail ~~beam~~ serves pumps A and C in SACS loop A, and the other serves pumps B and D in loop B. The monorails ~~beams~~ are located above elevation 102 feet in the reactor building. The top of each rail is at elevation 126 ft 10.75 in. Because dedicated SACS pumps hoists were not purchased, they will be borrowed from other locations when needed.

mm. SACS heat exchanger ~~rigging beam~~ hoist ~~(future)~~

Two parallel 2-ton capacity monorails ~~rigging beams~~ at one end of each SACS heat exchanger are designed to accommodate hoists for removal of the heat exchanger end covers. One set of monorails ~~beams~~ serves both SACS loop A exchangers, and the other set serves the loop B exchangers. The monorails ~~beams~~ are located above elevation 102 feet in the reactor building. The top of each rail is at elevation 127 ft 1.5 in. Because dedicated SACS heat exchanger hoists were not purchased, they will be borrowed from other locations when needed.

nn. Recombiner system hoists (00H318, 10H318)

These 1-1/2 (00H318) and 2-1/2 (10H318) ton capacity chain-operated monorail hoists are located above elevation 67 feet 3 inches in the service and radwaste area of the auxiliary building. Each hoist removes the valve operator from one of the control valves in the feed lines to the offgas recombiners, carries it to the hatch in the valve cell, and lowers it to a maintenance cart in the the access corridor at elevation 54 feet. Each valve operator weighs 943 pounds.

9.1.5.3 Safety Evaluation

All of the OHLHS cranes are evaluated in Table 9.1-10 with respect to whether they carry heavy loads over safety-related equipment located under the load path or on the next lower elevation. Table 9.1-10 excludes from further evaluation those OHLHS cranes that have no safety-related equipment below their load paths or only handle loads lighter than 1200 pounds although their design capacity is greater.

Those OHLHS cranes not excluded in Table 9.1-10 are listed in Table 9.1-12 along with the loads they carry, the lifting device, if any, for each load, and the safety-related equipment beneath the load path. Hazard elimination criteria are applied to each load handling situation identified in Table 9.1-12 to determine if it can be excluded from further evaluation. All equipment hatch load handling situations are dealt with in compliance with the guidelines of NUREG-0612.

Application of the NUREG-0612 guidelines, the exclusion criteria in Table 9.1-10, and the hazard elimination criteria in Table 9.1-12 show that there are no remaining OHLHS for which heavy load drop might prevent safe shutdown or decay heat removal, cause unacceptable radioactivity release, or expose spent fuel. The safe load paths for the OHLHS load situations in Table 9.1-12 are presented on Figures 9.1-32 through 9.1-3~~0~~7.

9.1.5.3.1 Reactor Building Polar Crane

Figure 9.1-32 shows the load paths for this crane. The reactor building polar crane is the only one of the OHLHS cranes, that is physically capable of carrying heavy loads over irradiated fuel. Both the main and auxiliary hoists are single-failure proof. Trolley and bridge travel limit switches, plus a set of bridge stops on the rail and main trolley stops near the middle of the bridge, together ensure that the main hoist cannot travel over the fuel pool. Figure 9.1-31 shows the main hook exclusion area. The cask loading pit is outside the exclusion area and separate from the spent fuel pool. The spent fuel cask, therefore, can not accidentally drop into the spent fuel pool. The cask is moved directly between the hatch, the cask washdown area, and the cask loading pit on the refueling floor as shown on the load path drawing, Figure 9.1-32.

Some safety features of the polar crane design are discussed in Section 9.1.5.2.1. In addition, the crane is designed to Seismic Category I criteria so that either hoist will retain its load during and after a SSE. Manually engaged anti-derail devices on both trolleys secure the trolleys when not in use and prevent rolling during an earthquake. Flat plate earthquake restraints welded onto the bottom of the girder end ties transfer the seismic loads to the reactor building wall through the crane rail.

The single-failure proof aspects of the polar crane design include complete redundancy for the sheaves, ropes, reeving, reducing gears, holding brakes, and other load path components of both the main and auxiliary hoists.

Figure 9.1-30 illustrates the single-failure proof auxiliary hoist design. The load is supported by the hook and two shackles, one on either side of the hook. The two separate load paths from the hook and shackles extend through the four side plates up to two separate sheave pins. Each of the two plates on either side of the load block is designed to support the design load. The trunnion applies the hook load to all 4 plates. Each shackle applies the hook load to the two side plates on its side. The side plates transmit the load to the two sheave pins. Each pin holds a sheave that is reeved independently. The block housing includes two through-bars that are designed to catch the wire ropes and/or sheaves if a sheave or sheave pin fails. Each sheave is independently reeved to the hoist drum, where the ropes are dead-ended to the drum.

Table 9.1-13 presents a point-by-point comparison of the reactor building polar crane design with the criteria of NUREG-0554, Single-Failure Proof Cranes for Nuclear Power Plants.

9.1.5.3.2 Reactor Building Polar Crane Lifting Devices

Lifting devices used by the polar crane are listed in Table 9.1-12. The special lifting devices, as defined by NUREG-0612, are listed in Table 9.1-14 along with the ~~status of compliance with ANSI N14.6-1978 and the~~ design safety factors.

A single-failure proof spent fuel shipping cask lifting device and cask lift point design in accordance with the requirements of NUREG-0612 will be selected for HCGS.

A single-failure proof conventional sling selected in accordance with NUREG-0612, Section 5.1.6(1) is used to lift the fuel pool gates. The fuel pool gates are the only heavy loads which must routinely be carried over the fuel pool. There are two lift points on each fuel pool gate. They are designed with a minimum static factor of safety of 20 with respect to material ultimate strength. This satisfies the NUREG-0612, Section 5.1.6 requirement for a safety factor of 5.

INSERT 4

The fuel pool slot plug sling is a single-failure proof conventional sling ~~special lifting device designed~~ selected to meet the requirements of NUREG-0612, Section 5.1.6. Each fuel pool slot plug has a single lifting point designed with a minimum static factor of safety of 20 with respect to material ultimate strength. This satisfies the NUREG-0612, Section 5.1.6 requirement for a safety factor of 10.

Although the special lifting device for the dryer-separator pool plugs is single-failure proof, the lift points are not. The dryer-separator pool plugs each have four lift points designed with a minimum static factor of safety of 10 with respect to material yield strength. Although not in strict compliance with NUREG-0612, Paragraph 5.1.6(3)(a), which requires redundant points, each having a design safety factor with respect to ultimate strength of five times the maximum combined concurrent static and dynamic load, the design is conservative and satisfies the intent of NUREG-0612.

The special lifting device for the reactor well shield plugs is single-failure proof in accordance with NUREG-0612, Section 5.1.6, but the lift points are not. Each shield plug has four lift points to prevent uncontrolled lowering of the load, assuming a single lift point failure. Each lift point has a static design safety factor of 5 with respect to yield strength. Although not in strict compliance with NUREG-0612, Paragraph 5.1.6(3)(a), the design is conservative and satisfies the intent of NUREG-0612.

The dryer-separator pool plugs and reactor well shield plugs discussed above are not carried over the fuel pool, but are carried over the reactor vessel. They are only carried over the reactor vessel when both the drywell head and the RPV head are in place. A shield plug drop will not damage fuel or cause

INSERT 4

The fuel rack lifting fixture will be used for several non-routine heavy load lifts over the fuel pool. It is used for installing the spent fuel rack modules. As described in Section 9.1.2.2.2.2, a base capacity of 1078 spent fuel cells plus 30 multipurpose cavities will be installed for initial plant operation. The remaining capacity of 17 rack modules, providing an additional 2976 cells, will be installed during plant operation. The lifting fixture design factors of safety versus yield and ultimate strengths are provided in Table 9.1-14. These factors meet the criteria of paragraph 5.1.6(1)(a) of NUREG-0612 for a single-failure-proof single load path special lifting device.

The lifting eye of the fixture is connected to the crane hook by a sling arrangement. The slings are selected to meet the single-failure-proof criteria of Section 5.1.6(1)(b) of NUREG-0612. The four legs of the fixture each have a J-shaped plate at the bottom. The fixture legs are lowered through four of the empty cells of the rack module being lifted, moved horizontally a short distance, and raised to hook to the module base. The four J-shaped plates contact the underside of the module base when it is being lifted. This design eliminates the need for lifting eyes on the module. The weight of the module, together with the shape of the lifting fixture plates, provides assurance that the fixture is securely attached to the module during lifting.

Thus, because there are no lift points on the modules, and both the crane and lifting fixture are single-failure-proof, the modules will be installed with a single-failure-proof handling system.

The modules will be lifted with the main hoist of the polar crane. Limit switches and travel stops, described in Section 9.1.5.2.1.5, will be removed as necessary to permit the main hook to travel into the main hook exclusion area shown on Figure 9.1-31 when the modules are installed.

unacceptable water leakage from the reactor. This conclusion is based on the assumption that a plug drop could damage the drywell head and seal plate, but would have a less severe impact than a drywell or RPV head drop. In the highly unlikely event of a plug drop, the consequences would satisfy the four evaluation criteria of NUREG-0612, Section 5.1.

The drywell head is lifted by the RPV head strongback. It is carried over the reactor vessel while the RPV head is in place. A drywell head drop will not damage fuel or cause unacceptable water leakage from the reactor. This conclusion is based on the assumption that a drywell head drop would be less severe than a RPV head drop. Depending on orientation, a drywell head drop could damage the insulation support structure, rupture the RPV vent and head spray piping, damage the seal plate, and hit the RPV itself. But because the drywell head weighs about 2/3 as much as the RPV head, and because some of its kinetic energy would be absorbed by the insulation support structure and head piping before it strikes the RPV head, which is still in place, a drywell head drop would not cause fuel damage or unacceptable water leakage. In the highly unlikely event of a drywell head drop, the consequences would satisfy the four evaluation criteria of NUREG-0612, Section 5.1.

The RPV head strongback lifts the RPV head. The strongback design satisfies the guidelines of ANSI N14.6-1978, Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kilograms) or More for Nuclear Materials, in general. However, it does not explicitly comply as recommended by NUREG-0612, Section 5.1.1(4). Further, the design satisfies the minimum design safety factor of 5 with respect to the material ultimate strength requirement of Section 5.1.1(4), but not the single-failure proof criterion of Section 5.1.6(1)(a) for a design safety factor of 10.

Because the strongback is not single-failure proof, an RPV head drop onto the open reactor vessel has been analyzed. Results show that vessel and core integrity would be maintained within the guidelines criteria of NUREG-0612, Section 5.1. The effects would be less severe than those due to the fuel handling accident analyzed in Chapter 15. Damage to the vessel would not be severe enough to cause water leakage that uncovers the fuel.

The dryer-separator sling lifts the steam dryer and the moisture separator. The sling design satisfies the guidelines of ANSI N14.6-1978 in general, but does not explicitly comply as recommended by NUREG-0612, Section 5.1.1(4). The design ~~also~~ factors of safety versus yield and ultimate strengths are provided in Table 9.1-14. They are less than the values of 3 versus yield and 5 versus ultimate required by Section 5.1.1(4).

~~satisfies the safety factor of 5 requirement of Section 5.1.1(4), but not the single-failure proof requirement of 5.1.6(1)(a) for a safety factor of 10.~~

Because the sling is not single-failure proof, both a dryer drop and a separator drop have been analyzed. Results show that vessel and core integrity would be maintained within the guideline criteria of NUREG-0612, Section 5.1. Damage to the reactor vessel would not be severe enough to cause water leakage that uncovers the fuel.

The service platform sling lifts the RPV service platform. The sling design satisfies the guidelines of ANSI N14.6-1978 in general, but does not explicitly comply as recommended by NUREG-0612, Section 5.1.1(4). Insert 5 ~~Also, the design satisfies the safety factor of requirement of Section 5.1.1(4), but not the single-failure proof requirement of Section 5.1.6(1)(a) for a safety factor of 10.~~

Because the service platform sling is not single-failure proof, a service platform drop has been analyzed. Results show that vessel and core integrity would be maintained within the guideline criteria of NUREG-0612, Section 5.1.

The fuel pool jib cranes are carried over the reactor vessel when the RPV head is off, but only when the RPV service platform is in place on the RPV flange. A jib crane drop could damage fuel if it managed to cause structural failure of the service platform. A conventional sling, selected in accordance with NUREG-0612, Paragraph 5.1.6(1)(b)(ii), is used to lift the jib crane. The load used to select the sling is two times the sum of the maximum static plus dynamic load. The dynamic load is assumed to be 0.25W, where W equals the weight of the jib crane. The load used is, therefore, 2(W+0.25W). The jib crane design has a single lift point with a design safety factor of 10 times the maximum combined concurrent static and dynamic load with respect to material ultimate strength as required by NUREG-0612, Paragraph 5.1.6(3)(b). The jib crane handling system, therefore, meets the single-failure proof criteria of NUREG-0612, Section 5.1.6.

No other heavy loads will be carried over the open reactor vessel.

(Ref. p.9.1-90) INSERT 5

The design factors of safety versus yield and ultimate strengths are provided in Table 9.1-14. The factor versus yield is greater than the value of 3, and the factor versus ultimate is less than the value of 5 required by Section 5.1.1(4) of NUREG-0612.

the RPV head strongback

The RPV head insulation and its support structure is carried over the RPV when the head is on. It is lifted by ~~slings selected to meet the single-failure proof criteria of NUREG-0612, Section 5.1.6(1)~~. The support structure is lifted in two pieces. The lift points on each piece are designed to meet the single-failure proof criteria of NUREG-0612, Section 5.1.6(3)(a).

The other heavy loads carried over the RPV while the head is on are the RPV stud tensioner and the RPV head stud rack. They will not cause fuel damage or unacceptable leakage because the drop would be less severe than a drywell or RPV head drop.

All heavy loads that need not be carried over the reactor well are restricted from this area during refueling. Administrative procedures help to control safe movement of all heavy loads.

In summary, a load drop into the reactor well could not affect safe shutdown capability since the well is only open when the reactor is shut down. Decay heat removal capability could be threatened only by a load large enough to damage the seal plate. Failure of the seal plate would not allow the large, heavy loads to fall into the drywell because their size is greater than the space between the RPV and the drywell. The reactor well and the drywell are lined with steel plate which will retain any concrete which is fragmented by swinging or falling loads. It is doubtful that other debris large enough to damage shutdown cooling piping could fall through the labyrinth of intervening piping and structural steel, including the massive primary containment radial box beams. The RHR shutdown cooling subsystem described in Section 5.4.7 includes a single suction line from reactor recirculation loop B. Therefore, a load drop into the reactor well could disable the shutdown cooling function of the RHR system. The design basis for this event is that any debris that managed to fall and disable RHR shutdown cooling would not have enough residual energy when it reached the components of this subsystem to do sufficient damage to prevent manual restoration of the cooling function. Damage such as a severed or crimped pipe, or complete loss of function of a suction line valve operator is not considered credible. Shutdown cooling would be manually restored as described in Section 5.4.7.1.5. If manual restoration cannot be achieved, an alternate flow path as described in Section 15.2.9 could be used. Similarly, if debris from the load drop were able to cause leakage from exposed reactor vessel piping, makeup water could be supplied by any of a number of RHR and core spray injection lines until the leak could be repaired. Therefore, the drop of a heavy load into the reactor well would not affect decay heat removal capability.

The flux monitor shipping crate is carried over the refueling floor by slings selected to meet the single-failure proof guidelines of NUREG-0612, Paragraph 5.1.6(1)(b).

Heavy loads carried over the refueling floor that employ lifting devices or lift points that are not single-failure proof weigh up to 107.5 tons.

These loads include the items listed below and are also tabulated, with their weights, in Table 9.1-12.

a. RPV head

b. Drywell head

c. Reactor well plugs-curved, 4

d. Reactor well plugs-straight, 2

e. Dryer separator pool plug-curved

f. Dryer separator pool plugs-straight, 3

g. RPV service platform

h. RPV stud tensioner

i. RPV head stud rack

The RPV and drywell heads each have four lift points. The drywell head lift points meet the single-failure proof guidelines of NUREG-0612, Section 5.1.6. The heads are handled as close to the refueling floor as is practical. Both heads are lifted by the RPV head strongback. As described above for loads handled over the reactor, the head strongback is not single-failure proof. However, the design is conservative and the potential for a load drop is very small.

The reactor well and dryer separator pool plugs are handled as close to the refueling floor as is practical. As described above for loads handled over the reactor, the four lift points of each plug are not single-failure proof. However, the design is conservative and the potential for a load drop is very small.

The RPV service platform has three lift points. The platform is handled as close to the refueling floor as is practical. It is lifted by the service platform sling. As described above for loads handled over the reactor, the sling is not single-failure proof. However, the design is conservative and the potential for a load drop is very small.

INSERT

The RPV stud tensioner has four lift points. The tensioner is handled as close to the refueling floor as is practical. The stud tensioner lifting device consists of four slings supplied with the tensioner. ~~The design is conservative and the potential for a load drop is very small.~~

The RPV head stud rack has a single lifting point. The stud rack is handled as close to the refueling floor as is practical. The stud rack is lifted by a sling selected to meet the single-failure proof criteria of NUREG-0612, Section 5.1.6(1).

Because the polar crane main hoist is prevented from traveling over the fuel pool, as described in Section 9.1.5.3.1, a load drop would not damage the fuel pool, spent fuel racks, or spent fuel. The RPV service platform, stud tensioner, and head stud rack are light enough to be handled by the polar crane auxiliary hoist. The loads paths are administratively controlled to keep these loads out of the main hoist exclusion area, i.e., from over the fuel pool.

In summary, a load drop on the refueling floor of any of the loads normally carried over the floor by a nonsingle-failure proof overhead handling system would satisfy the four evaluation criteria of NUREG-0612, Section 5.1.

Table 9.1-15 presents a failure modes and effects analysis for the reactor building polar crane.

9.1.5.3.3 Other OHLHS Cranes

All plant OHLHS cranes, except the reactor building polar crane, are evaluated below. ~~They are listed in the same order as they appear in Table 9.1-10.~~ The equipment tag numbers are shown in parentheses.

Each motorized hoist includes one 125%-capacity mechanical and one 125%-capacity electrical brake that is automatically applied on loss of power. Each bridge drive includes a 125%-capacity brake that automatically sets upon loss of power. Each trolley includes one 100%-capacity electrical brake that automatically sets upon loss of power.

The cranes and hoists shown as seismically secured in Table 9.1-10 have positive restraints that prevent crane derailment or crane parts from falling during an earthquake. These cranes are designed so that their parts will remain in place under a seismic acceleration of 7g vertical and 7g horizontal. The design also includes locking devices for use when the cranes are parked.

a. Personnel air lock hoist (10H217)

INSERT 7

This crane's load path is shown on Figure 9.1-33. There is no safe shutdown equipment directly below the load path. A portion of the primary containment suppression pool is located below the load path on the next lower elevation. The personnel air lock is part of the primary containment pressure boundary. It is only moved when the reactor is shut down. The air lock lift height above the floor is administratively limited to less than 2 feet 6 inches. This is the calculated maximum allowable lift height. A load drop would not penetrate the floor if dropped from less than 2 feet 6 inches above it. Movement of the nine shield blocks in front of the personnel air lock is administratively limited to reactor shutdown. The calculated maximum allowable lift height for the shield blocks is 1 foot. When the upper seven shield blocks are moved, they are higher than this. Administrative procedures require that the removal cart be in position below these seven blocks before the blocks are moved. The cart would absorb some of the energy of a load drop. A major portion of the remaining energy would be absorbed as the load punched through the floor. The low velocity

(Ref. p.9.1-93) INSERT 6

The tensioner sling design factors of safety versus yield and ultimate strengths are provided in Table 9.1-14. The factors calculated for the maximum combined static and dynamic load, assuming the entire load is carried by only two of the four wire ropes, are greater than the values of 6 versus yield and 10 versus ultimate required by paragraph 5.1.6(1)(a) of NUREG-0612 for a single-failure-proof single load path special lifting device.

(Ref p.9.1-94) INSERT 7

The air lock strongback design factors of safety versus yield and ultimate strengths are provided in Table 9.1-14. They meet the safety factor requirement of paragraph 5.1.6(1)(a) of NUREG-0612 for a single-failure-proof single load path special lifting device.

impact on the suppression pool shell below would probably deform but not punch through it. If the dropped block managed to penetrate the upper suppression pool shell, the residual energy would almost certainly be dissipated by the internal hardware (piping and catwalk) and the water itself before the block ruptured the lower portion of the shell and caused any water loss. Because the reactor would be already shutdown at the time of a shield block drop, the suppression pool would not have to be available for decay heat removal. The residual heat removal (RHR) system, operating in the decay heat removal mode, would take suction from a reciculation loop, pump through a RHR heat exchanger and back to the reactor. Therefore, a load drop that caused suppression pool water loss would not prevent decay heat removal.

b. Reactor recirculation pump motor hoist (1AH201, 1BH201)

Figure 9.1-33 shows this hoist's load path. The hoist is only operated during reactor shutdown.

Dropping a motor during the short time it is raised and free hanging is unlikely. The load is positively attached to the hoist hook by the hook safety latch. No intermediate lifting device is required. The hook directly engages the shackle pin on the top of the motor. The motor cannot be raised more than 5 feet because of the space limitation. It is normally raised no more than 3 feet.

If the motor were dropped, it would hit the pump~~s~~ and probably damage its coupling, seals, shaft, and bearings. The motor mount and the pump casing and its supports would absorb most of the energy and thereby protect the pump suction line between the pump and its upstream isolation valve from severe damage. The shutdown cooling line required for decay heat removal originates from the recirculation loop B suction line only. A motor drop could not prevent decay heat removal because the line branches from the recirculation loop piping about 15 feet above and more than 20 feet to the side of the potential motor impact point.

c. Reactor water cleanup filter-demineralizer hoist (~~10H213~~ 1AH220, 1BH220)
There is no safe shutdown or decay heat removal equipment beneath the load path of these hoists or on the next lower elevation. ~~Figure 9.1-38 shows this hoist's load path. Cable trays that carry power and instrumentation cables associated with single FRVS channel are located adjacent to the load path at its south end.~~ Operation of this hoist is unlikely because the pressure precoat type reactor water cleanup filter demineralizer vessels are designed to operate for the life of the plant without undergoing maintenance. ~~In the unlikely event of damage to a cable tray that causes loss of a FRVS unit, the redundant FRVS units would still be available. The only safety-related equipment on the next lower elevation is a ventilation duct. The duct is part of the containment prepurge cleanup system.~~ It only operates prior to occupancy ~~of the drywell or torus. It does not operate during normal plant operation or during shutdown. It is not required for safe shutdown or decay heat removal.~~

d. HPCI pump and turbine hoists (1AH211, 1BH211)

Figure 9.1-34 shows this hoist's load path. The only safe shutdown or decay heat removal equipment located in the load path is associated with the HPCI system. There is no lower floor elevation. A load drop during plant operation that disables the HPCI system would not prevent safe shutdown because HPCI does not function during normal shutdown. It may not be necessary to shutdown the plant, provided the applicable requirements of the plant Technical Specifications are met. That is, HPCI can be inoperable for 14 days if other ECCS divisions are available. Otherwise, hot shutdown must be achieved within 12 hours, or cold shutdown within 24 hours.

e. RCIC pump and turbine hoist (10H212)

This hoist does not handle heavy loads,

~~Figure 9.1-34 shows the safe load path. The only safe shutdown or decay heat removal equipment located in the load path is associated with the RCIC system. There is no lower floor elevation. A load drop during plant operation that disables the RCIC system would not prevent safe shutdown because RCIC does not function during normal shutdown. It may not be necessary to~~

~~shut down the plant, provided the applicable requirements of the plant Technical Specifications are met. That is, RCIC can be inoperable for 14 days if other ECCS divisions are available. Otherwise, hot shutdown must be achieved within 12 hours, or cold shutdown within 24 hours.~~

f. Main Steam Tunnel Underhung Crane ~~Outboard MSIV hoist~~ (10H214, ~~&~~ 10H223)

Figure 9.1-35 shows the safe load path. The reactor will be shut down when this hoist is used. There is no decay heat removal equipment located in the load path. All of the equipment below the valves is associated with either the main steam or feedwater systems. If an ~~the~~ operator were dropped, it would hit one or more of the following items before it could hit the steam tunnel floor: its valve body; the pipe on either side of the valve body; one of the other three main steam pipes; one of the feedwater lines; restraint steel; structural steel; and miscellaneous small pipe and valves of the main steam drains system. ~~Because the operator weighs 2440 pounds, the drop height would be less than 10 feet, and the intervening steel would absorb most of the energy, it is deemed incredible that a dropped valve operator could~~ INSERT punch through the ~~steam tunnel~~ floor. If a dropped operator managed to cause spalling after striking the floor, the concrete could hit one or more of the pipes in this area, or the torus itself. The pipes are associated with nuclear boiler instrumentation, liquid radwaste, RCIC, reactor water cleanup, core spray, fire protection, HPCI and primary containment instrument gas. None of the equipment below the load path is required to remove reactor decay heat. Therefore, decay heat removal ability would not be affected by a load drop from this hoist.

g. Inboard MSIV hoist (10H2~~03~~19)

Figure 9.1-35 shows the safe load path. The reactor is shut down when this hoist is used. There is no decay heat removal equipment located in the load path. The tops of the drywell radial structural steel and drywell floor framing cross beams are located at elevation 100 feet, just below the main steam lines. All of the equipment above this structural steel network is associated with either the main steam, primary containment instrument gas, or breathing air systems.

None is required for decay heat removal. If the operator were dropped, it would hit its own valve body or steam line, or one of the three other main steam pipes before it could contact the structural steel below, unless it were dropped in the removal space between main steam lines A and D. It would hit the steel directly if it were dropped in the removal space. The steel would stop a dropped valve operator. It would not fall to the lower elevation (drywell floor). There is no decay heat removal equipment on this lower elevation.

h. CRD service ~~r gging beam~~ hoist ~~(future)~~

Figure 9.1-35 shows the safe load path. There is no safe shutdown or decay heat removal equipment in the load path. The torus is below the load path on the next lower elevation. It is doubtful whether a dropped load could punch through the elevation 102 feet floor. Most loads actually weigh less than the 1200-pound heavy load limit. All loads are carried as close to the floor as is practical.

The following piping is located above the torus on the next lower elevation under the load path:

1. 18-inch RHR pump A discharge

2. 20-inch RHR shutdown cooling suction

3. 14-inch HPCI pump discharge

4. 12-inch HPCI turbine steam supply.

Three 1-inch channel A reactor vessel level, pressure and differential pressure instrument lines are also located in this area. If a dropped load during plant operation managed to penetrate the elevation 102 feet floor, or cause concrete spalling, and disable the shutdown cooling line, cold shutdown could still be achieved. As discussed in Section 15.2.9 for this situation, an alternate method to achieve and maintain cold shutdown that involves the safety/relief valves, (Ref. 9.1.5.3.3.f) INSERT 8

Because of the congested piping and massive restraint steel beneath the load path it is nearly impossible for a dropped valve operator to reach the steam tunnel floor. Together the congestion and energy absorbing capability make it certain that a dropped operator will not that the impact could cause water loss. However, water loss would not prevent decay heat removal.

jj. Solid radwaste monorail (00H316)

The hoist is remotely controlled with the aid of closed-circuit television from the drum-handling control panel located in the radwaste control room. If the hoist becomes inoperable, a mechanical retrieval device permits removal and/or repair as necessary, while keeping operator exposure as low as reasonably achievable.

There is no safe shutdown or decay heat removal equipment in the load path or on the next lower floor elevation. The drop of a drum could require implementation of isolation and decontamination procedures, but could not affect safe shutdown of the plant.

kk. Solid radwaste bridge crane (00H317)

The hoist is remotely controlled with the aid of closed-circuit television from the drum-handling control panel located in the radwaste control room. Independent motors control low and high speed crane movement. Eyelets on the bridge provide attachment points for a winch-type retrieval hoist in the event of a loss of crane electrical power.

There is no safe shutdown or decay heat removal equipment in the load path or on the next lower floor elevation. The drop of a drum could require implementation of isolation and decontamination procedures, but could not affect safe shutdown of the plant.

ll. SACS pumps ~~rigging beam~~ hoist ~~(future)~~

One monorail ~~rigging beam~~ serves the two pumps associated with safety auxiliaries cooling system (SACS) loop A, and the other serves the two pumps associated with loop B. A pump motor (6160 pounds) is only removed when the SACS coooling

{The heaviest anticipated maintenance load is the upper half of the pump casing (825 pounds), which is not a heavy load.

(physically separated)

loop associated with that pump is shutdown and completely isolated from the other (redundant) loop. This is not a normal maintenance lift. It would be done infrequently, if at all. The ~~rigging beam~~ monorail restricts the load path so that a load drop could only disable a pump or other equipment associated with the down loop.

A dropped motor would not punch through the elevation 102 feet floor because the deformation of the motor shroud, the intermediate pipe restraint steel, and the floor strength would absorb the kinetic energy of the dropped load. ~~A SACS pump motor weighs 1155 pounds.~~

mm. SACS heat exchanger ~~rigging beam~~ hoist ~~(future)~~

Two hoists, one mounted on each ~~rigging beam~~ monorail, work in tandem to remove a SACS heat exchanger return end cover. The configuration includes a separate sling and lifting point for each hoist. Each of the two hoist, sling, and lift point combinations is capable of independently supporting the cover. The OHLHS is thus single-failure proof in the sense that a single failure would not cause uncontrolled lowering of the load.

nn. Recombiner system hoists (00H318, 10H318)

This hoist does not handle heavy loads.

9.1.5.4 Inspection and Testing

9.1.5.4.1 Reactor Building Polar Crane

Final assembly and initial power operation of the bridge, both trolleys, and both hoists is done on site rather than in Paceco's shop. All crane parts subject to hoisting or seismic loads are nondestructively examined as described in Section 9.1.5.4.1.1.

The following steps are used to determine which items must be repaired or replaced after construction operation:

a. A review of maintenance logs to be aware of any crane operation difficulties and any special or unusual lifts that were accomplished during the construction program

b. A thorough visual inspection of all load bearing members

c. Crane is operated to clock speeds and motion smoothness

d. Maintenance personnel remove safety guards and access covers and clean the gears. Gears are then examined, relubricated, and replaced as necessary

e. Motor-coupling-reducer is checked for proper operation

f. Limit switches are checked for proper operation

g. Crane electrical control system is checked for proper sequencing and operation.

Preoperational tests of the polar crane include all of the specific heavy load handling operations that are performed during a normal refueling outage.

9.1.5.4.1.3 Operational Tests

In compliance with NUREG-0612, Section 5.1.1(6), the crane is inspected, tested, and maintained in accordance with Chapter 2-2 of ANSI B30.2-1976, Overhead and Gantry Cranes. ~~except when crane use frequency is less than the specified test or inspection frequency, the test or inspection is done prior to crane use.~~

9.1.5.4.2 Other OHLHS Cranes

Shop, preoperational, and operational tests on OHLHS cranes other than the polar crane are discussed in this subsection.

9.1.5.4.2.1 Shop Tests

All of the OHLHS cranes listed in Table 9.1-10, except items 1 (reactor building polar crane), 11 (turbine building bridge crane), 27 (solid radwaste monorail), 28 (solid radwaste bridge crane), and 38 (intake structure gantry crane) are functionally tested without load and at 150% of rated capacity. Each hoist brake is tested to confirm ability to brake the load from rated speed and hold it without slipping.

Shop testing of the reactor building polar crane is discussed in Section 9.1.5.4.1.1.

The turbine building crane is shop-assembled, except for the rope and blocks, to check fit. The trolley is powered along the bridge to check tracking. The hoist, trolley, and bridge drives are operated in the shop for 15 minutes.

The intake structure gantry crane is shop-tested at rated load. Each hoist brake is tested to confirm ability to brake the load from rated speed and hold it without slipping.

The solid radwaste monorail and solid radwaste bridge cranes are shop-tested at 125% of rated load.

9.1.5.4.2.2 Preoperational Tests

Each of the OHLHS cranes listed in Table 9.1-10 is given an operational performance test, a rated load test, and preoperational inspection in accordance with ANSI B30.2-1976, Chapter 2-2.

Preoperational testing of the reactor building polar crane is discussed further in Section 9.1.5.4.1.2.

After preoperational performance and rated load testing, per ANSI B30.2-1976, the turbine building bridge crane is operationally and rated load tested in accordance with Paragraph 1901.179(K) of OSHA. Each hoist brake is tested to confirm ability to brake the load from rated speed and hold it without slipping.

9.1.5.4.2.3 Operational Tests

All the OHLHS cranes listed in Table 9.1-10 that carry heavy loads over safety-related equipment (those not identified by exclusion criteria A, B, or C) are inspected, tested, and maintained in accordance with ANSI B30.2-1976. ~~An exception is when the crane use frequency is less than the specified test or inspection frequency, in which case the test or inspection is done before crane use.~~

9.1.5.5 Instrumentation

Instrumentation and controls for the reactor building polar crane are described in Sections 9.1.5.2.1 and 9.1.5.3.1 and Table 9.1-13. Supplemental information is presented below in Section 9.1.5.5.1.

9.1.5.5.1 Reactor Building Polar Crane

Bridge and trolley controls are the variable speed, reversing, magnetic, five-step type. Cab control handles are deadman-type with spring return. Hoist controls are A.C. static stepless-type in accordance with NEMA Industrial Control Standard ICS-3-442 Class III and OSHA. Release of a hoist controller stops the motion and sets the brakes.

The hoist control system limits lowering speed to 120% of full load hoist speed. Each hoist-holding brake system includes and overspeed switch that stops the motor and applies the brakes at 120% of maximum no load hoist speed. The hoists limit hook movement when starting from a standstill to 1/32 inch for the main hook and 5/16 inch for the auxiliary hook in either the hoist or lower direction.

Simultaneous motion of the bridge, trolleys, and hoists is possible whether control is from the cab or the pendant. Cab control includes a maintained contact, master on-off switch. Cab control is not possible unless the pendant is stored in its full up position. All pendant controls are momentary contact return to off pushbuttons. A deadman foot switch must be held down during crane operation from the cab.

For both the main and auxiliary hoists, a rotary limit switch coupled to the drum trips at the normal up and extreme low hook position. A block-operated overhoist limit switch backs up the normal "up" limit switch by stopping the drive and setting the brakes. Hoist overload switches shut off hoist power and set the brakes if the design loads (150 or 10 tons) are exceeded.

End of travel limit switches stop the main and auxiliary trolleys and the bridge at their normal stop positions.

The bridge, trolley, and hoist motors include overtemperature protection.

9.1.5.5.2 Other OHLHS Cranes

All cranes include a drum overspeed system to automatically set the load brake when hoist drum speed exceeds motor synchronous speed. A phase-loss protection system automatically stops the hoist and sets the holding brakes when hoist power is lost.

The turbine building bridge crane (item 11 in Table 9.1-10) control system includes redundant 125%-capacity hoist holding brakes that are automatically applied upon loss of power. The 125% trolley and bridge brakes are also automatically applied upon loss of power. The design includes hoist raising or lowering overtravel limit switches. Bridge and trolley travel limit switches cut power at the bridge or trolley travel limits. All crane motion control switches and pushbuttons are momentary-contact-return-to-off type.

The intake structure gantry crane (item 36 in Table 9.1-10) design includes automatic application of the mechanical hoist load brake and electrical hoist holding brake upon loss of power. Trolley end of travel limit switches cut motor power when the travel limits are reached.

9.1.5.6 SRP Rule Review

In SRP Section 9.1.5, Acceptance Criterion 2 refers to Regulatory Guide 1.13, Position C.3, which requires that interlocks be provided to prevent cranes from passing over stored fuel when fuel handling is not in progress.

At HCGS, only the main hoist of the polar crane is physically prevented from traveling over the spent fuel pool. The auxiliary hoist has no travel restriction. Preventing its travel over the fuel pool is not an auxiliary hoist design basis. Instead, the alternative basis of a single-failure proof hoist described in Section 9.1.5.3.1 is used. No loads are required to be routinely handled over the fuel pool when fuel handling is not in progress. The fuel pool gates are the only heavy loads routinely handled over the pool when fuel handling is in progress. A single-failure proof handling system lifts the gates, and any other nonroutine heavy loads that must be carried over the spent fuel pool.

Acceptance Criterion 2 also refers to NUREG-0612, which, in Paragraph 5.1.1(1), states that load paths should be clearly marked on the floor in the areas where heavy loads are to be handled.

At HCGS, load paths are not painted on the floor. They are omitted to avoid possible operator confusion in areas such as the refueling floor where multiple paths would cross. The paths are defined in the specific load handling procedures and shown on equipment layout load path drawings that are incorporated in the procedures. Deviations from defined load paths require written alternative procedures approved by the plant safety review committee.

INSERT 7

Acceptance Criterion 2 also refers to ANS 57.1, which in Paragraph 6.2.1.1(a) requires that the auxiliary fuel-handling crane be provided with an underload interlock that is actuated upon a reduction in load while lowering, to prevent any further downward travel.

At HCGS, the polar crane auxiliary hoist functions as the auxiliary fuel-handling crane. It does not have an underload interlock since it was purchased before ANS 57.1 was issued. The fuel pool gates are the only heavy loads normally handled over the fuel pool. A single-failure proof handling system lifts the gates, and any other nonroutine heavy loads that must be carried over the spent fuel pool.

9.1.6 REFERENCES

9.1-1 C. L. Martin, Lattice Physics Method, NEDO-20913, General Electric, June 1975.

9.1-2 AISC Manual of Steel Construction

9.1-3 AGMA Gear Classification Manual

9.1-4 Aluminum Construction Manual, Aluminum Association

9.1-5 AWS D1.1, Structural Welding

9.1-6 NEMA MG-1, Motor and General Standards

9.1-7 National Electric Code

9.1-8 OSHA 1910.179

9.1-9 OSHA, Vol 37, No. 202, Part 191 ON

(Ref. 9.1.5.6) INSERT 9

Because opaque plastic sheets may be taped to the floor where the potential for radioactive contamination exists, polar crane load paths painted on the refueling floor (elevation 201 feet) may not be visible. The alternative method that is used at HCGS for the polar crane is to make a person other than the crane operator (i.e., a signalman) responsible for assuring that the load path is followed. The signalman inspects the load path before the lift to ensure that it is clear, reviews the specific load handling procedure before the lift, and provides direction to the crane operator to ensure that the prescribed path is followed. The specific load handling procedures clearly define the duties and responsibilities of the operator, the signalman, and any other members of the load handling party.

The appropriate polar crane load path is temporarily marked with rope or pylons to provide a visual reference for the operator. If it is not possible to temporarily mark the load path, permanent or temporary match marks are used to assist in positioning the bridge and/or trolley for the lift. The method of marking the load path is defined in each specific load handling procedure.

The reactor building polar crane is the only non-exempt cab-operated crane at HCGS. Other non-exempt cranes, except for the main steam tunnel underhung crane are simple hoists on monorails where the load path cannot vary. Most lifts are short lifts where movement is limited to one coordinate axis in addition to the vertical. For these non-exempt, non-cab-operated hoists the specific load handling procedures define whether a signalman is used and whether the load path will be marked.

As described in Section 9.1.5.2.2 f, each of the monorails for the main steam tunnel underhung crane is mounted on end trucks that provide the capability for load movement in both coordinate axes in addition to the vertical.

Item Number	Crane or Hoist	Tag Number	Building	Floor Elev (ft)	Equipment Loc Fig Number
1	Reactor building polar crane	10H200	Reactor	201	1.2-32
2	Personnel air lock hoist	10H217	Reactor	102	1.2-28
3	Recirculation pump motor hoist	1AH201 1BH201	Reactor	102 (Drywell)	1.2-28
4	Reactor water clean-up filter/ demineralizer hoist	1AH220 1BH220	Reactor	178-6	1.2-31
5	HPCI pump and turbine hoist	1AH211 1BH211	Reactor	54	1.2-26
6	RCIC pump and turbine hoist	10H212	Reactor	54	1.2-26
7	Main steam tunnel underhung crane ~~Outboard MSIV hoist~~	10H214 10H223	Reactor	102	1.2-28
8	Inboard MSIV hoist	10H219	Reactor	102 (Drywell)	1.2-28
9	Vacuum breaker valve removal hoist	10H207	Reactor	54 (Torus)	1.2-27
10	Main steam line relief valve removal hoist	10H202	Reactor	135-6 (Drywell)	1.2-29
11	Turbine building bridge crane	10H102	Turbine	137	1.2-16
12	Feedwater heater removal hoist	1AH103 1BH103	Turbine	102	1.2-14
13	H&V equipment removal hoist	10H104	Turbine	171	1.2-17
14	Motor-generator set hoist	0AH105 0BH105	Turbine	137	1.2.-16

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TABLE 9.1-10

EAD HEAVY LOAD HANDLING SYSTEMS DATA SUMMARY

Column Area	Capacity (tons)	Max Vert Lift (ft in)	Seismic Cat I	Design Standard(2)	Is Load Over Safety-Related(5) Equipment?	Is Load Over Safety-Related(5) Equipment on Next Lower Elev	Exclusion Criterion(1)
N-V:14R-23R	150 main 10 aux	129-0	Yes	a, b	Yes	Yes	None
P-R:20R-23R	30	16-3	No(3)	c, d	No	Yes	None
Sa-Q;19R-20R Sa-T;18.9-17R	24	12-0	No(3)	c d	Yes	Yes	None
R-Q;1~~7~~5R-17R	10	26-0 ~~22-6~~	No	~~c~~ c d	NO ~~Yes~~	NO ~~Yes~~	~~None~~ B
W-V:18R-21R	4	9-10	No(3)	c, d	Yes	NA	None
W-V;17R-18R	3	9-0	No(3)	c, d	Yes	NA	~~None~~ C
P-Q;17R-20R	2-1/2	16-0 (10HZ14) 35-3 (10HZ23)	No(3)	a, d (4)	Yes	Yes	None
Q-R;17R-20R	2	15-5	No(3)	d	Yes	Yes	None
N-V;15R-22R	2	7-0	No(3)	d	Yes	No	C
Q-T;17R-20R	1	32-2	No(3)	c, d	No	Yes	C
E-F;12-29	220 main 45 aux	72-3 main 122-0 aux	No	a, b	~~Yes~~ No	~~Yes~~ No	~~None~~ B
E-Fq;18-22	24	12-6	No	d	No	No	B
H-F;12-13	15	37-0	No	c, d	No	No	B
Eu-H;26-29	15	16-5	No	c, d	No	No	B

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Item Number	Crane or Hoist	Tag Number	Building	Floor Elev (ft)	Equipment Loc Fig Number
15	Secondary condensate pump hoist	10H106	Turbine	54	1.2-12
16	Reactor feed pump hoist	1AH107 1BH107 1CH107	Turbine	137	1.2-16
17	Water box removal hoist	10H109 10H110	Turbine	81	1.2-13
18	Steam packing exhauster hoist	10H115	Turbine	77	1.2-13
19	Turbine-generator auxiliary crane	00H100	Turbine	137	1.2-16
20	Steam jet air ejector hoist	1AH117 1BH117 1CH117 1DH117	Turbine	77	1.2-13
21	Water box removal hoist	10H111 10H112	Turbine	81	1.2-13
22	Chiller tube removal hoist	10H118	Turbine	171	1.2-17
23	Emergency air compressor hoist	10H114	Turbine	123	1.2-15
24	Main air compressor hoist	00H113 10H113	Turbine	123	1.2-15
25	Vacuum pump water cooler hoist	10H116	Turbine	77	1.2-13
26	Heating and cooling coil removal hoist	1AH119 1BH119	Turbine	171	1.2-17
27	Solid radwaste monorail	00H316	Service and radwaste	102	1.2-20
28	Solid radwaste bridge crane	00H317	Service and radwaste	126-6	1.2-21
29	Demineralizer removal hoist	00H302	Service and radwaste	102	1.2-21
30	Decontamination evaporator hoist	00H305	Service and radwaste	54	1.2-18

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olumn rea	Capacity (tons)	Max Vert Lift (ft in)	Seismic Cat I	Design Standard(2)	Is Load Over Safety-Related Equipment?	Is Load Over Safety-Related Equipment on Next Lower Elev	Exclusion Criterion
-F;12-13	15	12-9	No	c, d	No	No	B
-H;17-22	15	19-9	No	c, d	No	No	B
-Eg;17-23	12	17- 4	No(3)	d, e	No	No	B
-G;16-17	10	14-11	No(3)	c, d, e	No	No	B
g-Eq;13-26	10	26-1/2	No(3)	c, b, d(4)	~~Yes~~ No	~~Yes~~ No	~~None~~ B
-G;17-26	8	13-4	No(3)	d, e	No	No	B
p-Et;17-23	8	17-4	No(3)	d, e	No	No	B
-H;17-21	5	14-4	No(3)	d, e	No	No	B
-Fd;14-16	4	7-0	No(3)	d, e	No	No	B
-Fd;11-14	3	7-1	No(3)	d, e	No	No	B
-F;14-16	2	6-9	No(3)	d, e	No	No	B
-H;13-18	1-1/2	13-0	No	c, d	No	No	B
a-M;36.9-42.6	1-1/2 ton	13-0	No	c, e	No	No	B
a-Mo;44.2-45.4	7.5 ton	39-6	No	a, b	No	No	B
L-Mc 25.9-34.6	10 ton	21-8	No	c, d	No	~~Yes~~ No	~~None~~ B
J-K;30.3-33.1	7-1/2	21-3	No	c, d	No	NA	B

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Item Number	Crane or Hoist	Tag Number	Building	Floor Elev (ft)	Equipment Loc Fig Number	Co Ar
31	Equipment decontamination room hoist	OOH314	Service and radwaste	102	1.2-20	Ha
32	Machine shop underhung crane	OAH301 OBH301 OCH301 ODH301	Service and radwaste	102	1.2-20	Ha 13
33	Waste evaporator recirculation pump hoist	OOH309 OOH310	Service and radwaste	54	1.2-18	Ha
34	Waste evaporator hoist	OOH312 OOH313	Service and radwaste	87	1.2-19	Ha
35	Diesel generator underhung crane	1AH400 1BH400 1CH400 1DH400	Control and diesel generator	102	1.2-35	S-
36	Intake structure gantry crane	OOH500	Intake structure	122, 128	1.2-41	A
37	Personnel lock shield removal hoist	1AH218 1BH218	Reactor	102	1.2-28	P-
38	Recombiner system hoist	10H318 00H318	Control and diesel generator	67-3	1.2-24	B
39	CRD service hoist	(6)	Reactor	102	1.2-28	T-
40	SACS pumps hoist	(6)	Reactor	102	1.2-28	V-
41	SACS heat exchanger hoists	(6)	Reactor	102	1.2-28	V-

(1) Exclusion criteria:
- A. This crane is located in a building or structure that contains no safety-related or safe shutdown equipm
- B. This crane's load path does not pass over any safety-related or safe shutdown equipment on the floor bel
- C. Although this crane's capacity is greater than 1200 pounds, its dedicated load is lighter than 1200 poun

(2) Design standards:
- a. ANSI B30.2.0 Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)
- b. CMAA 70 Electric Overhead Traveling Cranes
- c. HMI 100 Electric Wire Rope Hoists
- d. ANSI B30.16 Overhead Hoists (underhung)
- e. ANSI B30.11 Monorail Systems and Underhung Cranes

(3) Seismically secured (designed so that all parts remain in place under 7g horizontal and vertical seismic accel restraints and locking devices).

(4) The design also uses ANSI B30.17 (Overhead and Gantry Cranes - Top Running Bridge, Single Girder, Underhung Hoist) as a guide.

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(5) For the purposes of this table safety related is defined as "required for plant shut

(6) This hoist will be borrowed from another location when needed. T

lumn ea	Capacity (tons)	Max Vert Lift (ft in)	Seismic Cat I	Design Standard(2)	Is Load Over Safety-Related Equipment?	Is Load Over ~~Safety-Related~~ stet Equipment on Next Lower Elev	Exclusion Criterion
-K;28.8-33.1	~~10~~ 5	10-2	No(3)	c, d, e(4)	No	No	B
-Md: .6-25.9	5	11-0	No	d, e(4)	No	No	B
-K;15.8-19.9	2	9-3	No(3)	d, e	No	NA	B
-K:15.8-19.9	1	9-4	No(3)	d, e	No	No	B
-W;24.3-34.6	2	19-6	No(3)	c, d, e	Yes	Yes	None
-C:5-9	30 main 15 aux	65'-0" main 88'-0" aux	No	b, d, e	Yes	Yes	None
-R;20R-22R	15	23-0	No(3)	~~b.~~ c, d	No	Yes	None
-K;34.6-39.1	2.5(10H318) 1.5(00H318)	18-8	No	d	No	No	C
W; 21R-22R	LATER	(6)	(6)	(6)	No	Yes	None
W; 15R-17R / 20R-22R	LATER	(6)	(6)	(6)	Yes	Yes	None
W; 13.6-15R ent. 22R-24.2	5 (6)	(6)	(6)	(6)	No	Yes	None

ow or on the next lower elevation.
ds.

erations, and equipped with positive

down or decay heat removal."

he monorail capacity is shown.

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TABLE 9.1-12

OHLHS LOADS OVER SAFETY-RELATED EQUIPMENT

Heavy Load		Load Weight	Lifting Device	Safe Load Path Fig	First Elevation: Feet	First Elevation: Safety-Related, Safe Shutdown, or Decay Heat Removal Equipment	First Elevation: Hazard Elimination Criterion[1]	Second Elevation: Feet	Second Elevation: Safety-Related, Safe Shutdown, or Decay Heat Removal Equipment	Second Elevation: Hazard Elimination Criterion
Crane/Hoist: Reactor Building Polar Crane (Item 1, Table 9.1-10)										
a.	Reactor well shield plugs	107-1/2 tons	Shield plug ~~strongback~~ sling	9.1-32	201	(2)(3)	a, f	162	(2)(3)	a, f
b.	Drywell head	65 tons	RPV head strongback	9.1-32	201	(2)(3)	a, e, f	162	(2)(3)	a, e, f
c.	Reactor vessel head	97 tons	RPV head strongback	9.1-32	201	(2)(3)	a, e, f	162	(2)(3)	a, e, f
d.	Moisture separator	73-1/4 tons	Dryer/separator sling	9.1-32	201	(2)(3)	a, e, f	162	(2)(3)	a, e, f
e.	Steam dryer	45 tons	Dryer/separator sling	9.1-32	201	(2)(3)	a, e, f	162	(2)(3)	a, e, f
f.	Dryer/separator pool plugs	90 tons	Pool plug ~~strongback~~ grapple	9.1-32	201	(2)(3)	a, f	162	(2)(3)	a, f
g.	Spent fuel shipping cask	110 tons	Fuel cask yoke	9.1-32	201	(2)(3)	a, d	162	(2)(3)	a, d
h.	Auxiliary hoist load block	1 ton	(None required)	9.1-32	201	(2)(3)	d	162	(2)(3)	d
i.	Main hoist load block	10 tons	(None required)	9.1-32	201	(2)(3)	a, d	162	(2)(3)	a, d
j.	Spent fuel pool slot plugs	9 tons	Single-failure proof sling	9.1-32	201	(2)(3)	d	162	(2)(3)	d
k.	Spent fuel pool gates	~~6~~ 3.4 tons	Single-failure proof sling	9.1-32	201	(2)(3)	d	162	(2)(3)	d

TABLE 9.1-12 (cont)

	Heavy Load	Load Weight	Lifting Device	Safe Load Path Fig	First Elevation: Feet	First Elevation: Safety-Related, Safe Shutdown, or Decay Heat Removal Equipment	First Elevation: Hazard Elimination Criterion(1)	Second Elevation: Feet	Second Elevation: Safety-Related, Safe Shutdown, or Decay Heat Removal Equipment	Second Elevation: Hazard Elimination Criterion
l.	RPV service platform	8 tons Std	Service platform sling	9.1-32	201	(2)(3)	e, f	162	(2)(3)	e, f
m.	Head stud rack	1.5 tons	Single-failure proof sling	9.1-32	201	(2)(3)	e, f	162	(2)(3)	e, f
n.	Vessel head insulation and frame	5 tons	RPV head strongback ~~Single-failure proof slings~~	9.1-32	201	(2)(3)	e, f	162	(2)(3)	e, f
o.	Flux monitor shipping crate	2.5 tons	Single-failure proof slings	9.1-32	201	(2)(3)	d	162	(2)(3)	d
p.	Stud tensioner frame	5.3 ~~5 tons~~	RPV stud tensioner sling	9.1-32	201	(2)(3)	e, f	162	(2)(3)	e, f
q.	Head strongback	4.4 ~~5~~ tons	(None required)	9.1-32	201	(2)(3)	a, e, f	162	(2)(3)	a, e, f
r.	Spent fuel cask yoke	6 tons	(None required)	9.1-32	201	(2)(3)	a, d	162	(2)(3)	a, d
s.	Hatch cover 8' x 4'	2.4 tons	Single-failure proof slings	9.1-32	201	(2)(3)	f	162	(2)(3)	f
t.	Hatch cover 10' x 10'	7.5 tons	Single-failure proof slings	9.1-32	201	(2)(3)	f	162	(2)(3)	f
u.	Refueling bellows guard ring	10 tons	Single-failure proof sling	9.1-32	201	(2)(3)	e, f	162	(2)(3)	e, f
v.	Jib crane	1.6 tons ~~3200 lb~~	Single-failure proof sling	9.1-32	201	(2)(3)	d	162	(2)(3)	d
w.	Channel handling boom crane	0.8 ton ~~1600 lb~~	(None required)	9.1-32	201	(2)(3)	e, ~~f~~ d	162	(2)(3)	e, ~~f~~ d
x.	Dryer-Separator sling	2 tons	(None required)	9.1-32	201	[LATER]	d	162	[LATER]	d
y.	Spent fuel rack	10 tons	Fuel rack lifting fixture	9.1-32	201	[LATER]	d	162	[LATER]	d

TABLE 9.1-12 (cont'

Heavy Load	Load Weight	Lifting Device	Safe Load Path Fig	First Elevation: Feet	First Elevation: Safety-Related, Safe Shutdown, or Decay Heat Removal Equipment	First Elevation: Hazard Elimination Criterion(1)	Second Elevation: Feet	Second Elevation: Safety-Related, Safe Shutdown, or Decay Heat Removal Equipment	Second Elevation: Hazard Elimination Criterion
3. Fuel rack lifting fixture	1.1 tons	(None required)	9.1-32	201	[illegible]	d	162	[LATER]	d
aa. Reactor well shield plug sling	4.5 tons	(None required)	9.1-32	[illegible]	[illegible]	d	162	[illegible]	d
[illegible]. [illegible]-separator pool plug grapple	6 tons	(None required)	9.1-32	201	[illegible]	d	162	[LATER]	d

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Heavy Load	Load Weight	Lifting Device	Safe Load Path Fig	First Elevation: Feet	First Elevation: Safety-Related, Safe Shutdown, or Decay Heat Removal Equipment	First Elevation: Hazard Elimination Criterion[a]	Second Elevation: Feet	Second Elevation: Safety-Related, Safe Shutdown, or Decay Heat Removal Equipment	Second Elevation: Hazard Elimination Criterion
Crane/Hoist: Personnel Air Lock Hoist (Item 2, Table 9.1-10)									
a. Air lock	30 tons	Air lock strongback	9.1-33	102	None	b	77	Torus, and core spray HPCI, and SRV discharge piping	b, c
b. Upper shield block	21 tons	None required	9.1-33	102	None	b	77	Torus and core spray, HPCI, and MSRV discharge piping	b, c
c. Lower shield blocks (8)	17 tons	None required	9.1-33	102	None	b	77	Torus and core spray, HPCI and MSRV discharge piping	b, c
Crane/Hoist: Recirculation Pump Motor Hoist (Item 3, Table 9.1-10)									
Recirculation pump motor	24 tons	None required	9.1-33	102 (inside drywell	Recirculation (1AP201, 1BP201) and associated piping and conduit	b, c	87 (bottom of drywell)	None	-
Crane/Hoist: Reactor Water Cleanup Filter/Demin. Hoist (Item 4, Table 9.1-10)									
RWCS filter/demineraliser cell cover shield blocks	8 tons	Conventional slings	9.1-38	178'6"	Class 1E cable trays	b	162	Ventilation duct	c
Crane/Hoist: HPCI Pump and Turbine Hoist (Item 5, Table 9.1-10)									
HPCI pump and turbine parts (turbine case)	3.75 tons	Conventional slings	9.1-34	54	Pumps (10P204, 10P217) turbine (10S211) & HPCI piping	b, c	(No lower elevation)		-

TABLE 9.1-12 (cont)

Heavy Load	Load Weight	Lifting Device	Safe Load Path Fig	First Elevation: Feet	First Elevation: Safety-Related, Safe Shutdown, or Decay Heat Removal Equipment	First Elevation: Hazard Elimination Criterion[(a)]	Second Elevation: Feet	Second Elevation: Safety-Related, Safe Shutdown, or Decay Heat Removal Equipment	Second Elevation: Hazard Elimination Criterion
Crane/Hoist: RCIC Pump and Turbine Hoist (Item 6, Table 9.1-10)									
RCIC pump and turbine parts (turbine case)	2.35 tons	Conventional slings	9.1-34	54	Pump (10P203) turbine (10S12) & RCIC piping	b, c	(No lower elevation)		-
Crane/Hoist: ~~Outboard MSIV Hoist~~ Main Steam Tunnel Underhung Crane (Item 7, Table 9.1-10)									
Valve operators: Main steam isolation valve ~~operators~~	~~2440 lb~~ 1.8 tons	Conventional slings	9.1-35	102	MSIVs (HV F028A-D), main steam, and feedwater piping	c	54	Torus plus core spray, containment instrument gas, RCIC and HPCI piping, and nuclear boiler system instrumentation	b, c
Main steam stop valve	0.9 tons								
M.O. feedwater check valve	0.9 tons								
Crane/Hoist: Inboard MSIV Hoist (Item 8, Table 9.1-10)									
Main steam isolation valve operators	~~2440 lb~~ 1.8 tons	Conventional slings	9.1-35	102	MSIVs (HV F022 A-D) and main steam piping	c	87 (bottom of drywell)	Main steam, containment instrument gas and breathing air piping	c, e
Crane/Hoist: Turbine Building Bridge Crane (Item 11, Table 9.1-10)									
Turbine-generator rotor	202 tons	Conventional slings	Later	137	RPS (reactor protection system) conduit	b	102	RPS conduit	b
Crane/Hoist: Turbine Generator Auxiliary Crane (Item ~~20~~ 19, Table 9.1-10)									
Outer shield	6.75 tons	Conventional slings	Later	137	RPS conduit	b	102	RPS conduit	b
Crane/Hoist: Demineralizer Removal Hoist (Item 29, Table 9.1-10)									
Liquid radwaste filter bundle	1800 lb	Conventional slings	9.1-39	102	None	-	87	RPS conduit	b

TABLE 9.1-12 (cont)

Heavy Load	Load Weight	Lifting Device	Safe Load Path Fig	First Elevation: Feet	First Elevation: Safety-Related, Safe Shutdown, or Decay Heat Removal Equipment	First Elevation: Hazard Elimination Criterion[(1)]	Second Elevation: Feet	Second Elevation: Safety-Related, Safe Shutdown, or Decay Heat Removal Equipment	Second Elevation: Hazard Elimination Criterion
Crane/Hoist: Diesel Generator Underhung Crane (Item 35, Table 9.1-10)									
Diesel generator parts, e.g., combustion air cooling water heat exchanger tube bundle	3540 lb	Conventional slings	9.1-36	102	Diesel generators (1AG400-1DG400) and assoc cooling piping	b, c	77	Associated cooling piping	b, c
Crane/Hoist: Intake Structure Gantry Crane (Item 36, Table 9.1-10)									
Traveling screen, S.W. pump, and misc equipment	19 tons	Conventional slings	9.1-37	123	Screens (S501) & heaters (VE507) & S.W. pumps (P502)	b, c	93	Strainers (F509)	b, c
Crane/Hoist: Reactor Building Personnel Lock Shield Removal Hoist (Item 37, Table 9.1-10)									
T-shaped shield block	21 tons	None required	9.1-33	102	None	-	54	Torus and core spray, HPCI, and SRV discharge piping	b, c
Crane/Hoist: CRD Service Hoist (Item 39, Table 9.1-10)									
CRD maintenance equipment	1 ton (maximum)	conventional slings	9.1-35	102	None	-	77	[Later]	b, e

TABLE 9.1-12 (con

Heavy Load	Load Weight (tons)	Lifting Device	Safe Load Path Fig	Feet	First Elevation Safety-Related, Safe Shutdown, or Decay Heat Removal Equipment	Hazard Elimination Criterion(1)	Feet	Second Elevation Safety-Related, Safe Shutdown, or Decay Heat Removal Equipment	Hazard Elimination Criterion
Crane/Hoist: SACS Pumps A and C Hoist (Item 40, Table 9.1-10)									
Motor	3.1	conventional sling	9.1-35	102	SACS Loop A pumps, remaining motor, associated piping	b	77	SACS Loop B piping (TACS & diesels supply & return)	LATER
Crane/Hoist: SACS Pumps B and D Hoist (Item 40, Table 9.1-10)									
Motor	3.1	conventional sling	9.1-35	102	SACS Loop B pump, remaining motor, associated piping	b	77	None	–
Crane/Hoist: SACS Heat Exchanger A Hoists (Item 41, Table 9.1-10)									
return end cover	9.2	conventional slings	9.1-35	102	None	–	77	Later	Later
Crane/Hoist: SACS Heat Exchanger B Hoists (Item 41, Table 9.1-10)									
return end cover	9.2	conventional slings	9.1-35	102	None	–	77	Later	Later

(1) Hazard elimination criteria:

a. Crane travel for this area/load combination is prohibited by electrical interlocks or mechanical stops.

b. System redundancy and separation precludes the loss of the capability of the system to perform its safety-related function following this load drop in this area.

c. Site-specific considerations, such as maintenance sequencing, eliminate the need to consider this load/equipment combination.

d. The likelihood of a handling system failure for this load is extremely small; i.e., Section 5.1.6 of NUREG-0612 is satisfied, the OHS is single-failure-proof.

e. Analysis demonstrates that crane failure and load drop will not prevent safe shutdown or decay heat removal, or cause unacceptable radiation release.

f. The likelihood of a handling system failure is small. The system design meets the intent of NUREG-0612 (not dropping the load).

(2) Irradiated fuel.

(3) Reactor vessel.

TABLE 9.1-13

REACTOR BUILDING POLAR CRANE DESIGN COMPARISON
WITH NUREG 0554, SINGLE FAILURE PROOF
CRANES FOR NUCLEAR POWER PLANTS
(MAY 1979)

NUREG Section	Complies	Does Not Comply	Notes
1. INTRODUCTION	X		
2. SPECIFICATION AND DESIGN CRITERIA			
2.1 Construction and Operating Periods	X		(1)
2.2 Maximum Critical Load	X		(2)
2.3 Operating Environment	X		(3)
2.4 Material Properties	X		(4)
2.5 Seismic Design	X		(5)
2.6 Lamellar Tearing	X		(6)
2.7 Structural Fatigue	X		(7)
2.8 Welding Procedures	X		
3. SAFETY FEATURES			
3.1 General	X		
3.2 Auxiliary Systems	X		
3.3 Electric Control System	X		(8)
3.4 Emergency Repairs	X		
4. HOISTING			
4.1 Reeving System	X		(9)
4.2 Drum Support	X		(10)
4.3 Head and Load Blocks	X		
4.4 Hoisting Speed	X		
4.5 Design Against Two-Blocking	X		(11)

NUREG Section	Complies	Does Not Comply	Notes
9. OPERATING MANUAL	X		
10. QUALITY ASSURANCE	X		(17)

Notes:

(1) Section 2.1 - The load lifts during construction were not greater than those for plant operation; therefore, no separate specifications were prepared.

(2) Section 2.2 - The reactor building polar crane main hoist is designed to handle a maximum critical load (MCL) of 130 tons. The MCL rating will be clearly marked on the main hoist. The design rated load (DRL) of 150 tons provides an overall increase of 15% in the crane's load handling ability above its MCL capacity to compensate for wear and exposure.

The reactor building polar crane auxiliary hoist is designed to handle a MCL of 8.7 tons. The MCL rating will be clearly marked on the auxiliary hoist. The design rated load (DRL) of 10 tons provides an overall increase of 15% in the crane's load handling ability above its MCL capacity to compensate for wear and exposure.

(3) Section 2.3 - All identified parameters, except maximum rate of pressure increase and emergency corrosive conditions, were specified. A maximum rate of pressure increase was not specified because it was judged not significant to safe design of the crane. Because it is in the reactor building, outside the drywell, the crane would not be subjected to the high accident pressure (62 psig) possible inside the drywell. The maximum pressure increase specified for crane design is -.25 in. wg minimum to +7 in. wg maximum. Emergency corrosive conditions were not specified because none were identified that would prevent safe crane operation.

(4) Section 2.4 - The minimum specified operating temperature is 60°F. Materials for structural members essential to structural integrity are impact-tested unless exempted by the provisions of Paragraph AM-218 of the ASME Code, Section VIII, Division 2. All structural members, except the main hoist drums, are exempt under Paragraph AM-218.2, which withdraws the impact test requirement if stress intensity is less than 6000 psi. The main hoist drums are Charpy-tested

stainless steel auxiliary hoist wire ropes, with independent wire rope center, are 1 inch in diameter with an ultimate breaking strength of 77,200 pounds each.

(10) Section 4.2 - The main hoist and auxiliary hoist drum assemblies, each with its shafts and bearings, are designed at factors of safety not less than 10. Safety lugs are provided inside each trolley truck to sustain the drum assembly hubs in the event of drum shaft failure at either end. Upper sheave shafts and block swivel assemblies are provided with safety retainers and block housings capable of sustaining the load in case of shaft or swivel failure. Drum movement in this event is mechanically limited so that the gears and holding brakes remain engaged.

(11) Section 4.5 - Dual upper limit switches of diverse design in series, and an overload cutoff switch on each hoist stop the hoist motor and set the brakes. Motor overtemperature switches activate warning lights in the cab and on the pendant. Each limit switch allows the hoist motor to be operated in reverse after it has opened.

(12) Section 6.1 - An emergency breaker switch located at the refueling floor level cuts power to the crane independently of the crane controls.

(13) Section 6.2 - The crane ~~is~~ does not lift spent fuel assemblies. ~~assemblies~~

(14) Section 6.4 - Jogging and plugging are considered in the crane controls design. Drift point is not provided for bridge or trolley movement.

(15) Section 6.6 - Manual controls for hoisting and trolley movement are not provided on the trolley. Manual controls for the bridge are not located on the bridge.

(16) Section 8.3 - The crane design does not include an energy controlling device between the load and head blocks. Therefore, the two-block test is not done. Instead, the two-block test consists of verification that the two uptravel limit switches on each hoist function as designed.

(17) The crane is procured under a QA program that complies with the applicable provisions of ANSI N45.2-1971. Field installation, testing, operator qualification, and crane operation comply with ANSI B30.2.

per ASTM A 370. The crane was not subjected to coldproof testing because low alloy steel, such as ASTM A 514, is not used. Cast iron is not used for any crane parts.

(5) Section 2.5 - The SSE design vertical acceleration is less than 1g. Therefore the bridge and trolley wheels will not jump up off their tracks during a seismic event. The bridge and trolley designs include horizontal seismic restraints that would prevent the wheels from leaving the tracks.

(6) Section 2.6 - Nondestructive examination (NDE) was done on all welds whose failure could cause a drop of a critical load. Section 9.1.5.4.1.1 describes the NDE in more detail. Lamellar tearing of these welds is not expected to occur.

(7) Section 2.7 - A structural fatigue analysis was not part of the design requirements for the reactor building polar crane. The crane is classified as a low-use crane according to the guidelines of CMAA Specification 70. Structural fatigue analysis is not considered necessary in view of the low number of load cycles expected.

(8) Section 3.3 - Cab controls are deadman-type with spring return. A deadman foot switch in the cab must be held down during crane operation. Release of the switch will stop the crane and set the brakes. Overspeed switches on the hoist drives stop the motors and set the brakes at 120% of no load speed. Pendant controls are momentary contact pushbuttons that return to off when released. Pendant control includes an emergency stop pushbutton that stops power to all drivers.

(9) Section 4.1 - The maximum fleet angle from drum to lead sheave in the load block or between individual sheaves does not exceed 3-1/2 degrees at any one point during hoisting. Reverse bends are not used in the reeving system. Each main hoist rope is reeved through block and upper sheave assemblies so that its eight parts provide two parts in each quadrant of the load block about the vertical axis of the hook. With both ropes effective, the load is supported by sixteen parts at an effective static factor of safety of 10. If one rope loses its effectiveness, the load is supported by the eight parts of the remaining rope at a static factor of safety of ~~5~~ 6. The extra improved plow steel main hoist wire ropes, with independent wire rope center are 1-1/2 inches in diameter with an ultimate breaking strength of 228,000 pounds each. With both auxiliary hoist ropes effective, the load is supported by four parts at an effective static factor of safety of 15. If one rope loses its effectiveness, the load is supported by two parts of the remaining rope at a static factor of safety of 5. The

HCGS

Table 9.1-

HOPE CREEK SPECIAL LIFTING DEVICE

Item	Special Lifting Device	Rated Capacity tons	Maximum Load Weight tons	Lifting Device Weight tons	Maximum Static Load tons
1.	RPV head strongback	100	97	4.4	101.4
2.	Dryer separator sling	73.5	73.3	2	75.3
3.	RPV service platform sling	7.2	5.9	0.1	6
4.	Fuel cask yoke	110	110	(2)	(2)
5.	Reactor well shield plug sling	107.5	107.5	4.5	112
6.	Dryer separator pool plug grapple	38	38	6	44
7.	DELETED				
8	RPV stud tensioner sling	5.3	5.3	0.2	5.5
9.	Personnel air lock strongback	30	30	0.7	30.7
10.	Fuel rack lifting fixture	10	10	1.1	11.1

Notes:

(1) Deleted

(2) The spent fuel shipping cask and yoke are not yet known for HCGS. The ca rail cask. A 110-ton load is used here to acknowledge assum development of the NAC 12/32 cask to replace the proje

(3) Deleted

(4) Dynamic load factor = 0.15 ≤ 0.005 (HOIST SPEED, feet per minute)

(5) NUREG-0612, Section 5.1.1(4) requires a factor of safety of 3 load. The static load — includes the weight of the load pl

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FACTORS OF SAFETY

Dynamic Load Factor(4)	Maximum Combined Static and Dynamic Load tons	Stress Design Factor vs. Yield(5)	Stress Design Factor vs. Ultimate(5)	NUREG-0612 Section 5.1.1(4) Factor of Safety Compliance
0.15	116.6	3.5	5.7	Yes
0.15	86.6	2.6	4.3	No
0.175	7.1	3.0	3.1	NO
0.15	(2)	(2)	(2)	(2)
0.15	128.8	3	5	Yes
0.15	50.6	6	10	Yes
0.175	6.5	14.2	14.7	Yes
0.15	35.3	6	10	Yes
0.175	13.0	6	10	Yes

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k weight and dimensions used for the HCGS design are based on the projected 125-ton NLI 20/32 ption of the NLI cask design responsibility by Nuclear Assurance Corporation (NAC) and subsequent ted NLI cask.

≤ 0.50.

versus yield and 5 versus ultimate strength for the combined static and dynamic s the weight of the special lifting device.

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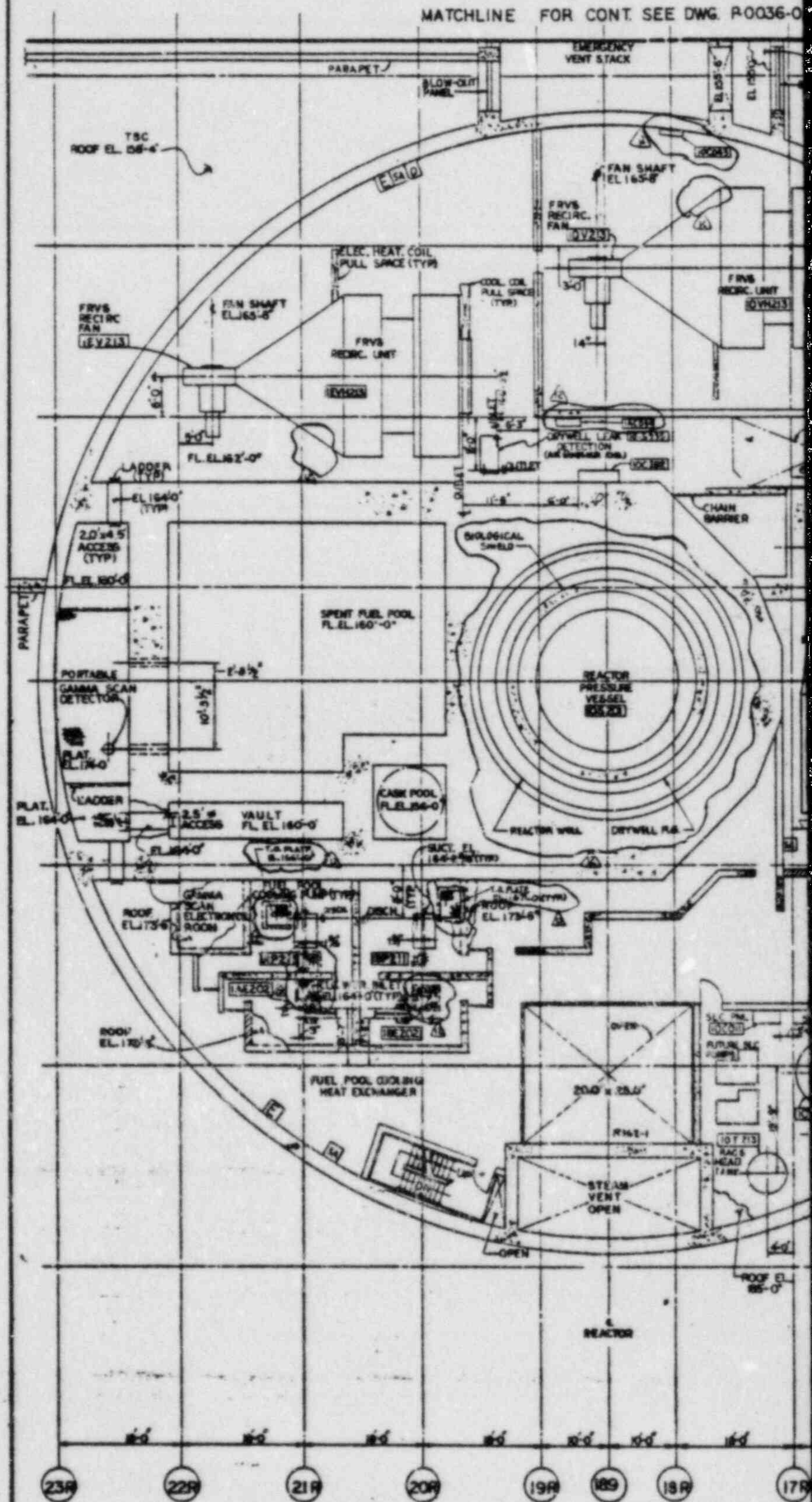


PLAN AT EL. 162'-0"

8 7 6

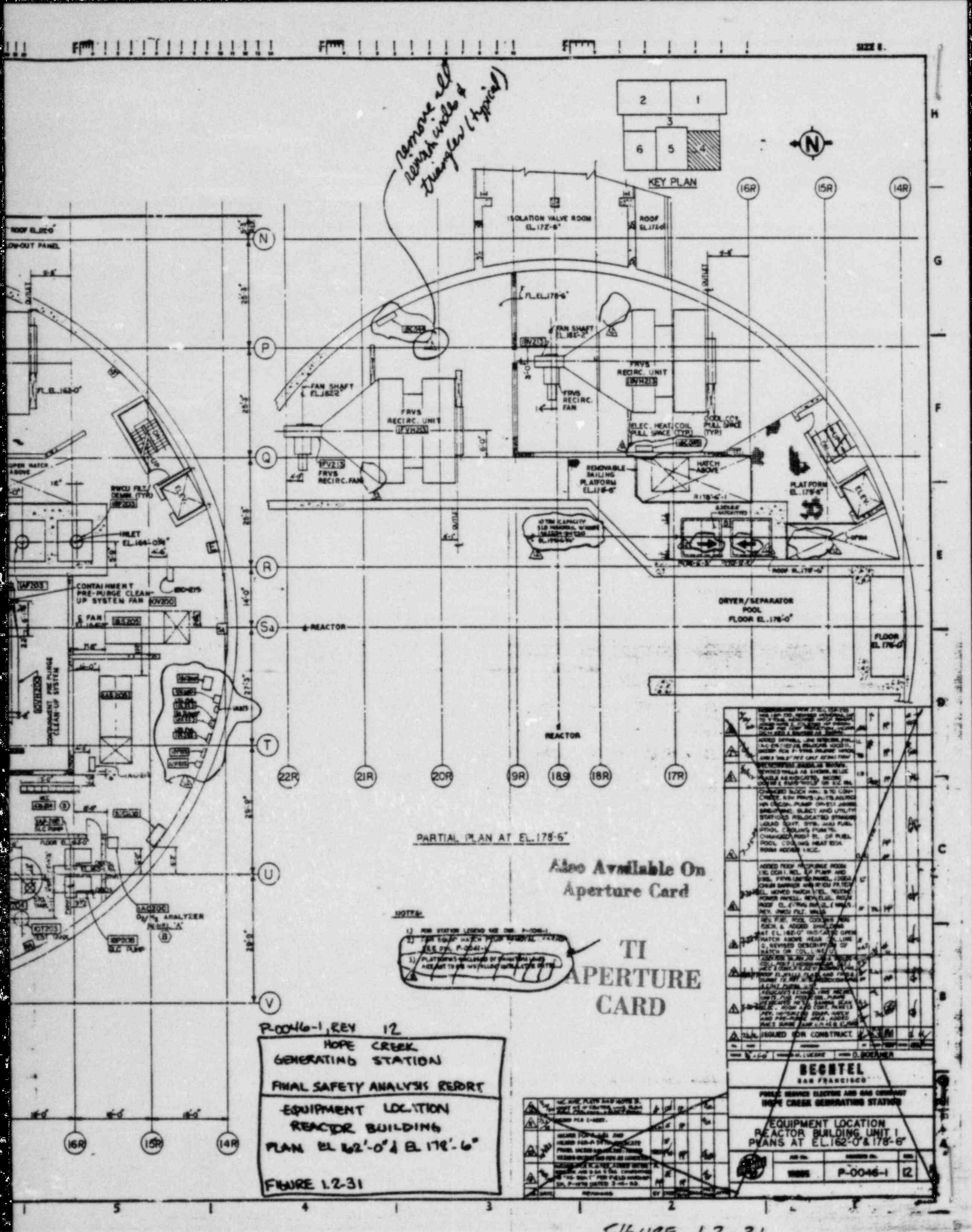
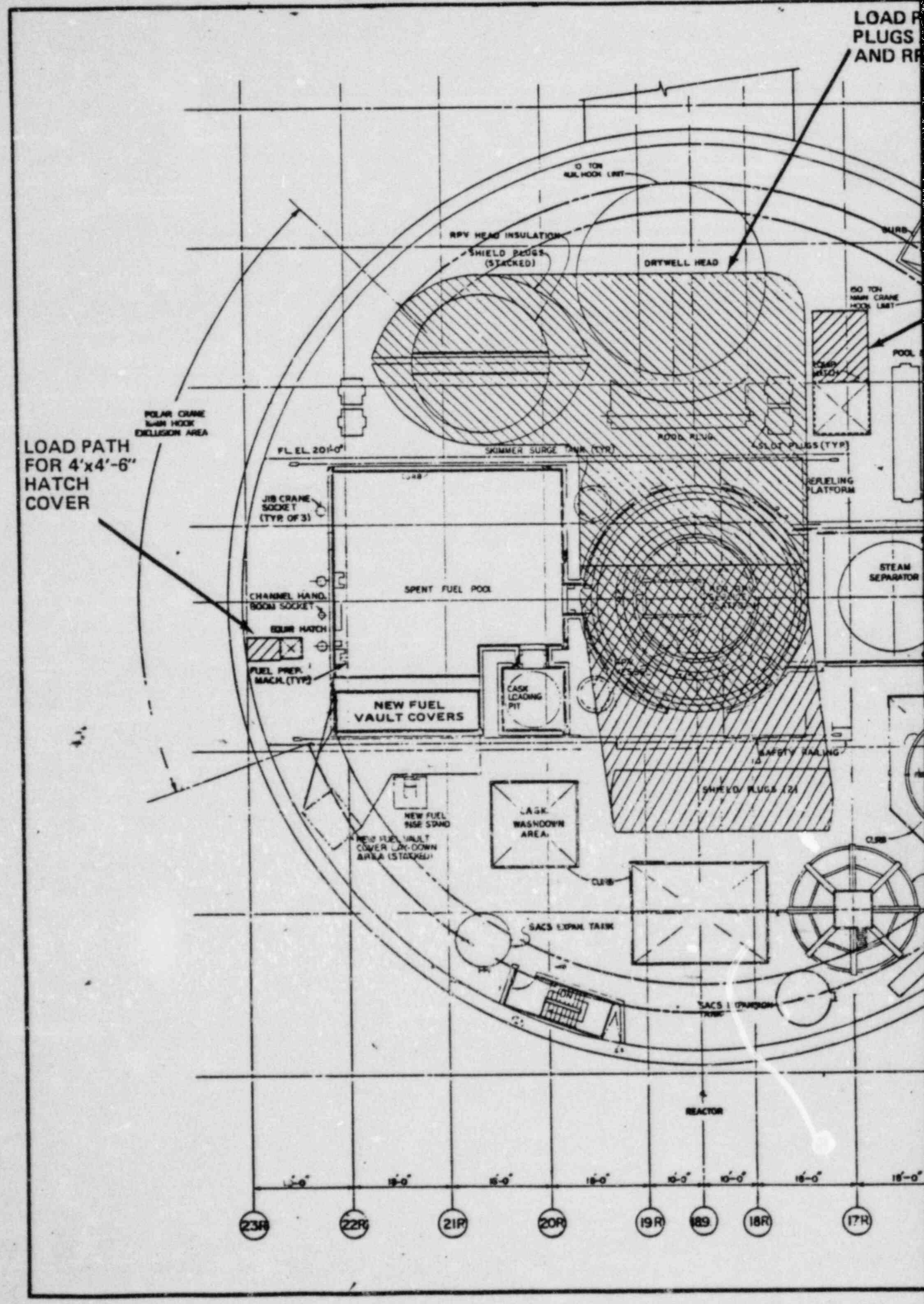
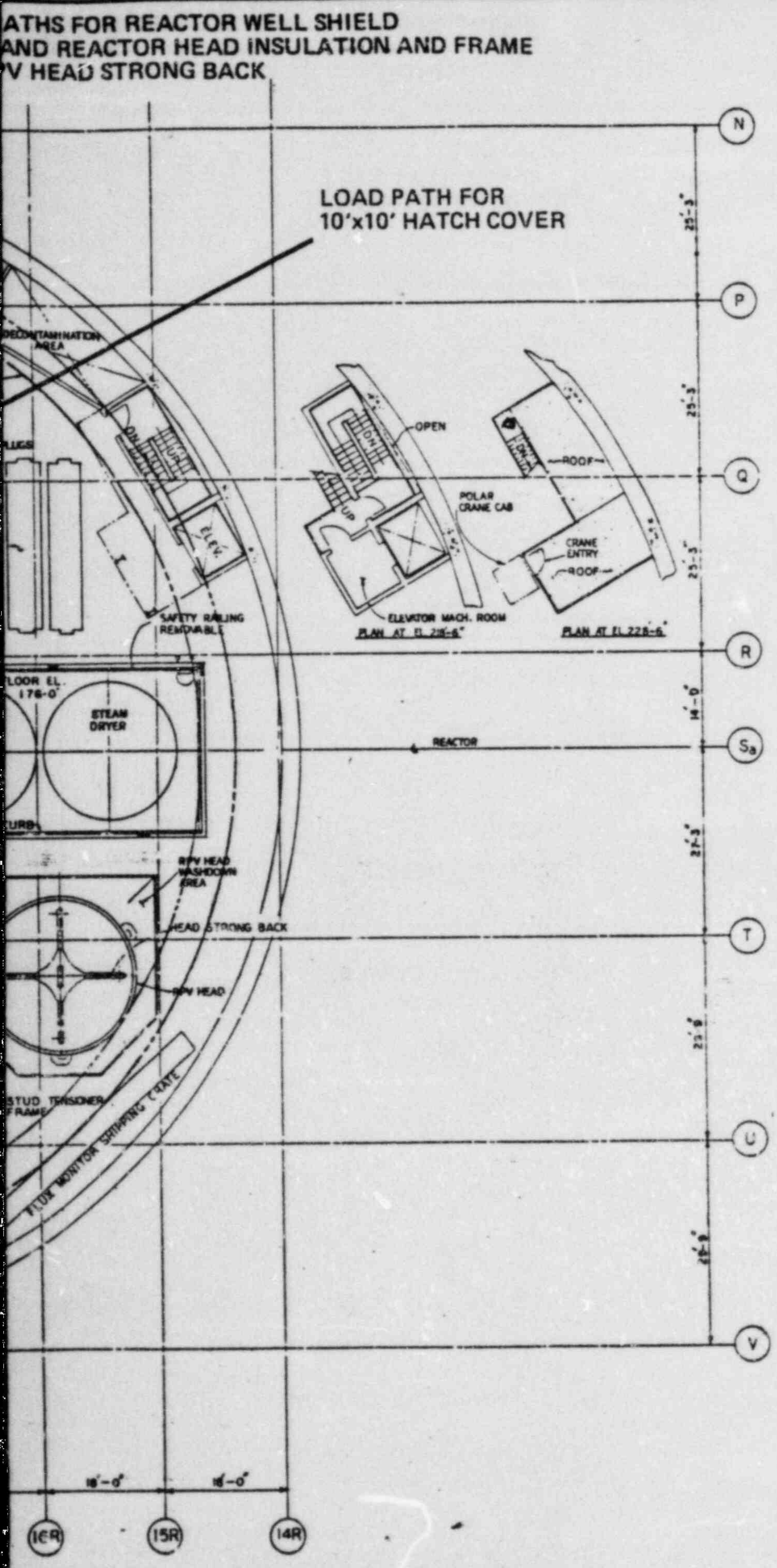


FIGURE 1.2-31

8409110175-05



LOAD PATH
FOR 4'x4'-6"
HATCH
COVER
POLAR CRANE
MAIN HOOK
EXCLUSION AREA
RPV HEAD INSULATION
SHIELD PLUGS
(STACKED)
DRYWELL HEAD
150 TON
MAIN CRANE
HOOK LIMIT
FL. EL. 201'-0"
SKIMMER SURGE TANK (TYP)
SLOT PLUGS (TYP)
REFUELING
PLATFORM
JIB CRANE
SOCKET
(TYP OF 3)
SPENT FUEL POOL
STEAM
SEPARATOR
CHANNEL HAND.
BOOM SOCKET
FUEL PREP.
MACH. (TYP)
NEW FUEL
VAULT COVERS
CASK
LOADING
PIT
SAFETY RAILING
SHIELD PLUGS (2)
NEW FUEL
INSP. STAND
NEW FUEL VAULT
COVER LAY-DOWN
AREA (STACKED)
CASK
WASHDOWN
AREA
CURB
SACS EXPAN. TANK
REACTOR
19'-0"
18'-0"
10'-0"
23R
22R
21R
20R
19R
18R
17R



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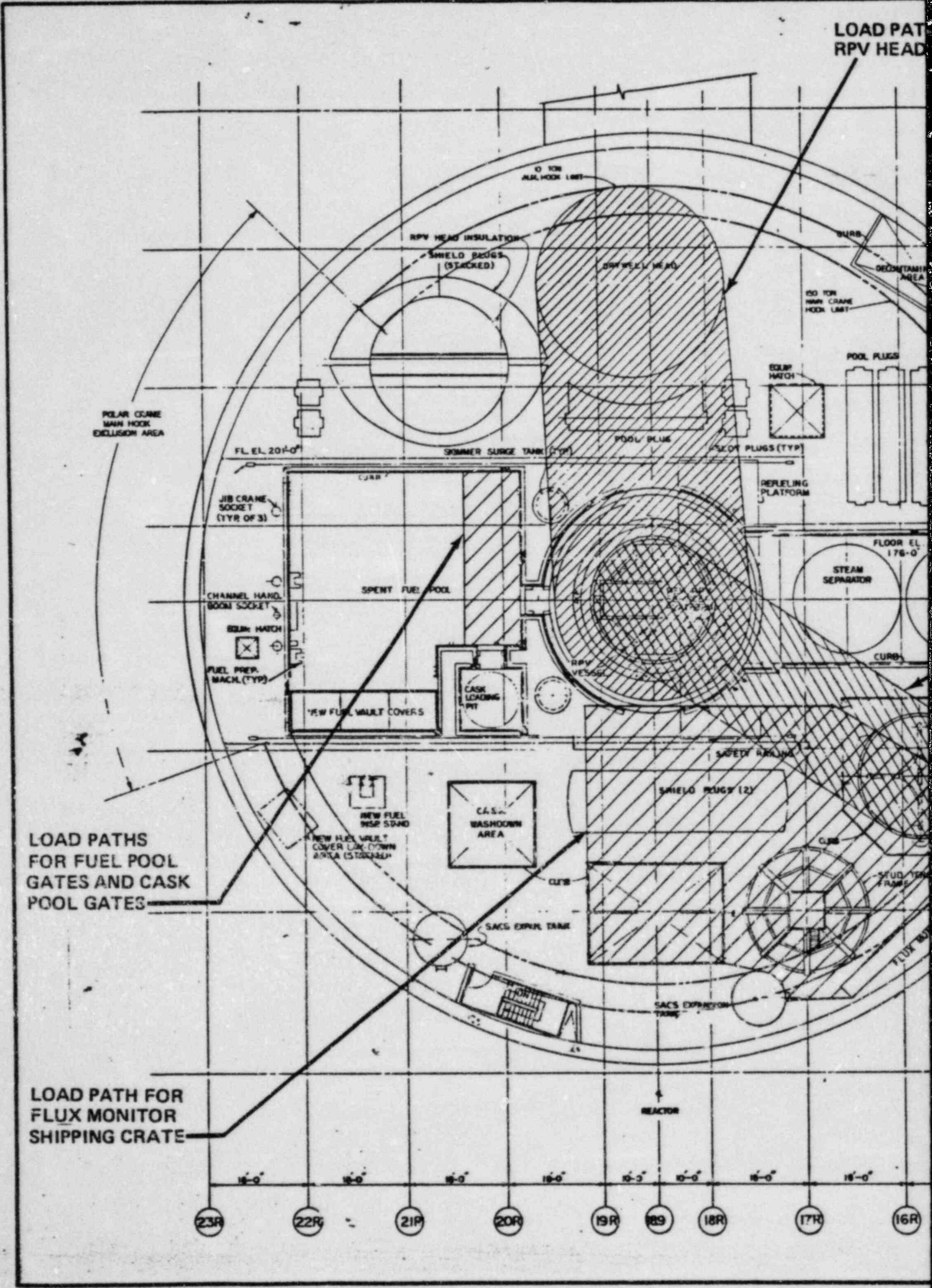
FINAL SAFETY ANALYSIS REPORT

SAFE LOAD PATH DRAWING (EL. 201)
REACTOR BUILDING POLAR CRANE
10H200

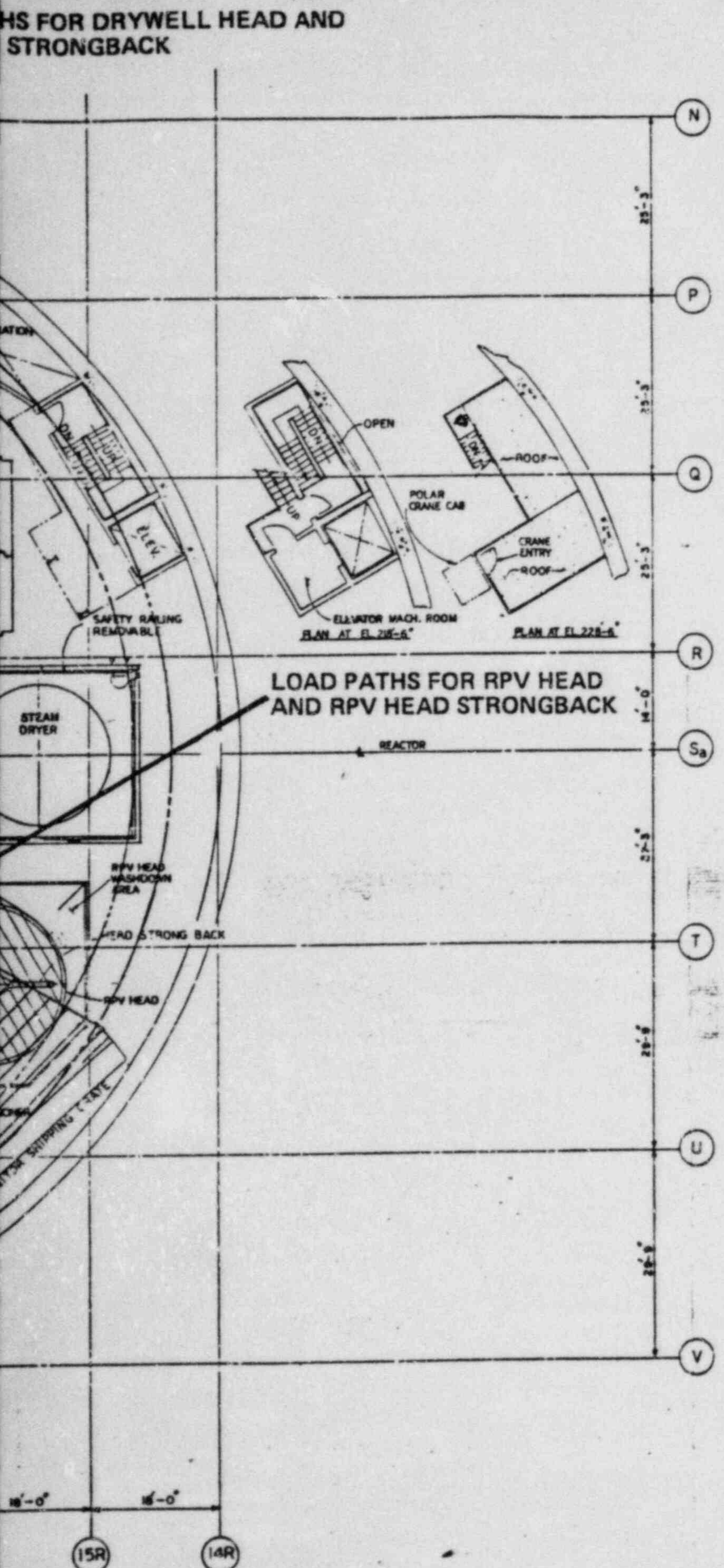
FIGURE 9.1-32
SHEET 1 OF 12

~~AMENDMENT 8, 10/84~~

8409110175-06



LOAD PATH FOR RPV HEAD
LOAD PATHS FOR FUEL POOL GATES AND CASK POOL GATES
LOAD PATH FOR FLUX MONITOR SHIPPING CRATE
RPV HEAD INSULATION
SHIELD PLUGS (STACKED)
DRYWELL HEAD
POLAR CRANE MAIN HOOK EXCLUSION AREA
FL. EL. 201'-0"
SKIMMER SURGE TANK (TYP)
POOL PLUG
SLOT PLUGS (TYP)
REFUELING PLATFORM
POOL PLUGS
EQUIP. HATCH
CURB
JIB CRANE SOCKET (TYP OF 3)
SPENT FUEL POOL
CHANNEL HAND. BOOM SOCKET
FUEL PREP. MACH. (TYP)
NEW FUEL VAULT COVERS
CASK LOADING PIT
RPV VESSEL
FLOOR EL. 176'-0"
STEAM SEPARATOR
SAFETY RAILING
SHIELD PLUGS (2)
NEW FUEL INSP. STAND
CASK WASHDOWN AREA
SACS EXPN. TANK
SACS EXPANSION TANK
REACTOR
23R
22R
21R
20R
19R
18R
17R
16R



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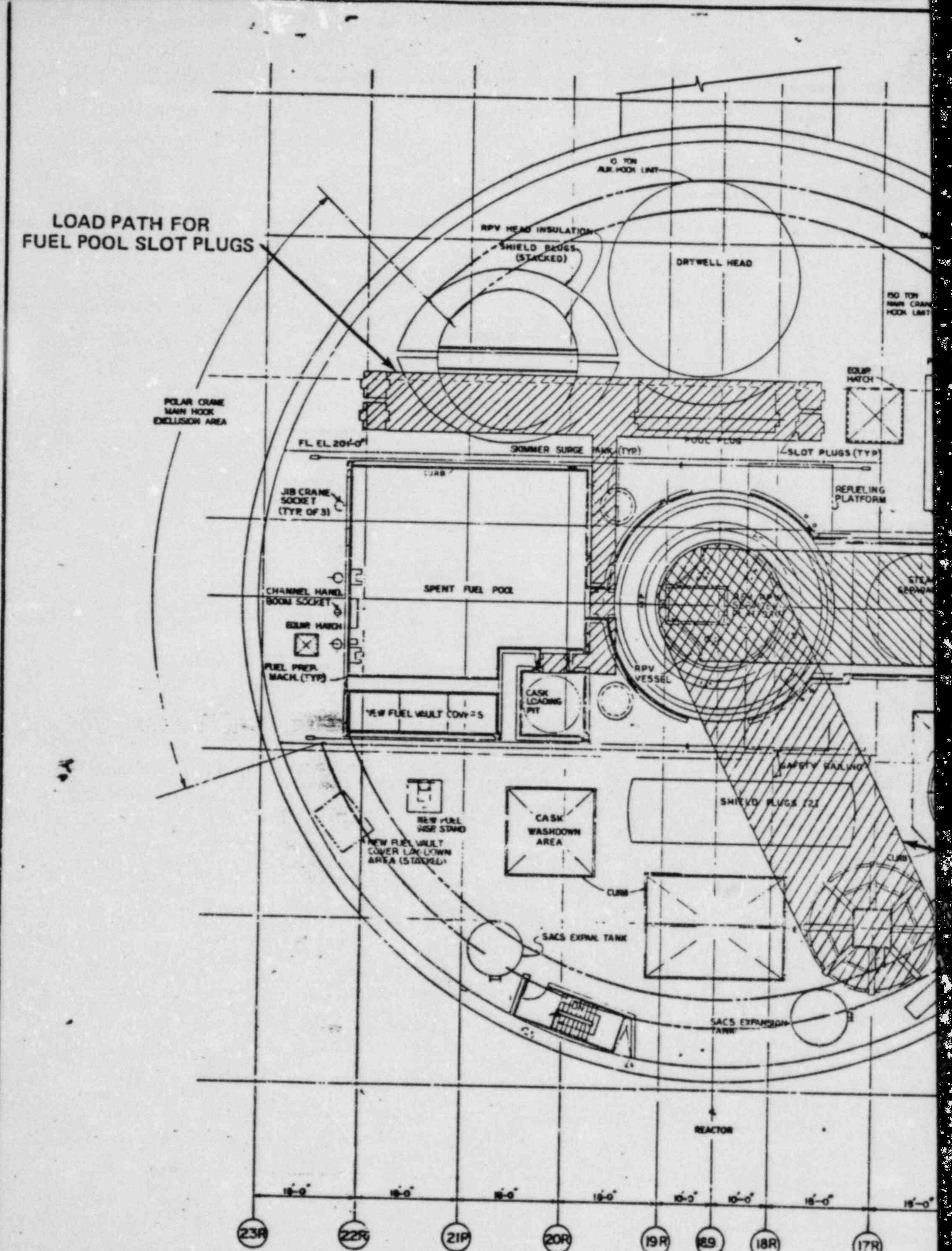
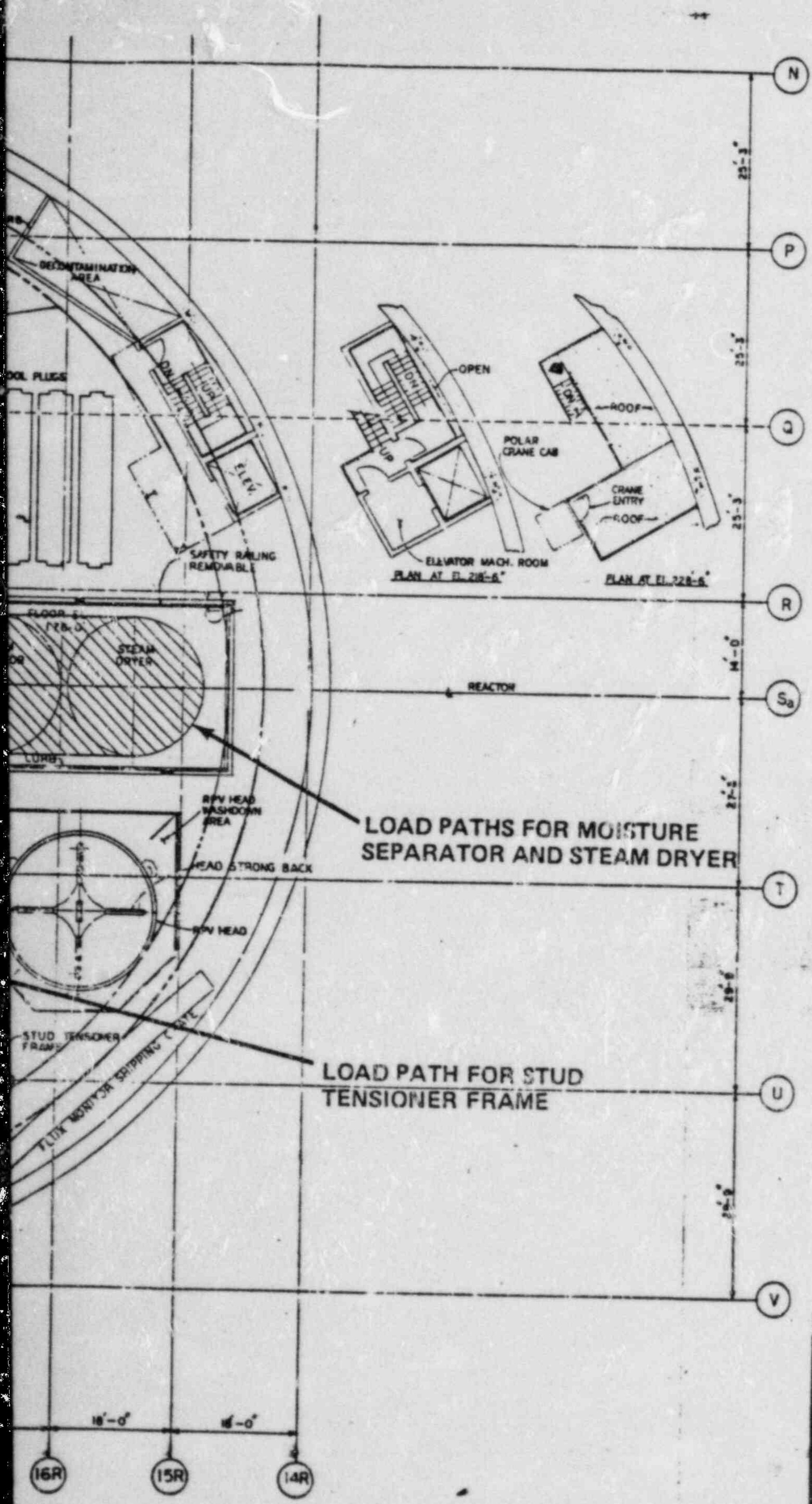
SAFE LOAD PATH DRAWING (EL. 201)
REACTOR BUILDING POLAR CRANE
10H200

FIGURE 9.1-32
SHEET 2 OF 12

~~AMENDMENT 8, 10/84~~

8409110175-07


LOAD PATH FOR
FUEL POOL SLOT PLUGS
10 TON
AUX HOOK LIMIT
RPV HEAD INSULATION
SHIELD PLUGS
(STACKED)
DRYWELL HEAD
POLAR CRANE
MAIN HOOK
EXCLUSION AREA
EQUIP HATCH
FL. EL. 201'-0"
SKIMMER SURGE TANK (TYP)
POOL PLUG
SLOT PLUGS (TYP)
REFUELING
PLATFORM
JIB CRANE
SOCKET
(TYP. OF 3)
CHANNEL HAND.
BOOM SOCKET
SPENT FUEL POOL
EQUIP. HATCH
FUEL PREP.
MACH. (TYP)
CASK
LOADING
PIT
RPV
VESSEL
SAFETY RAILING
SHIELD PLUGS
NEW FUEL
INSP. STAND
CASK
WASHDOWN
AREA
NEW FUEL VAULT
COVER LAY-DOWN
AREA (STACKED)
CURB
SACS EXPAN. TANK
SACS EXPANSION
TANK
REACTOR
23R
22R
21P
20R
19R
18.9
18R
17R



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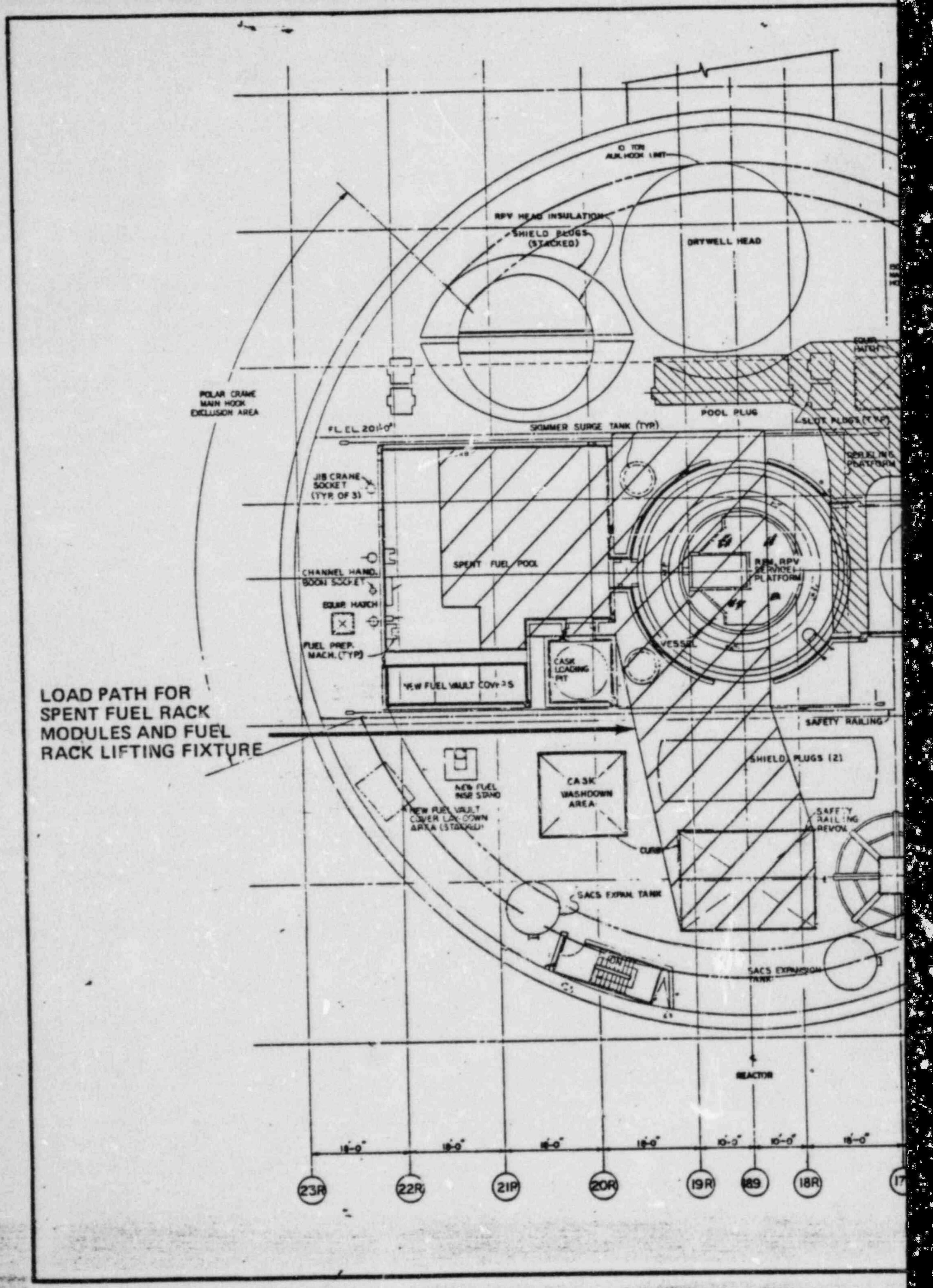
FINAL SAFETY ANALYSIS REPORT

SAFE LOAD PATH DRAWING (EL. 201)
REACTOR BUILDING POLAR CRANE
10H200

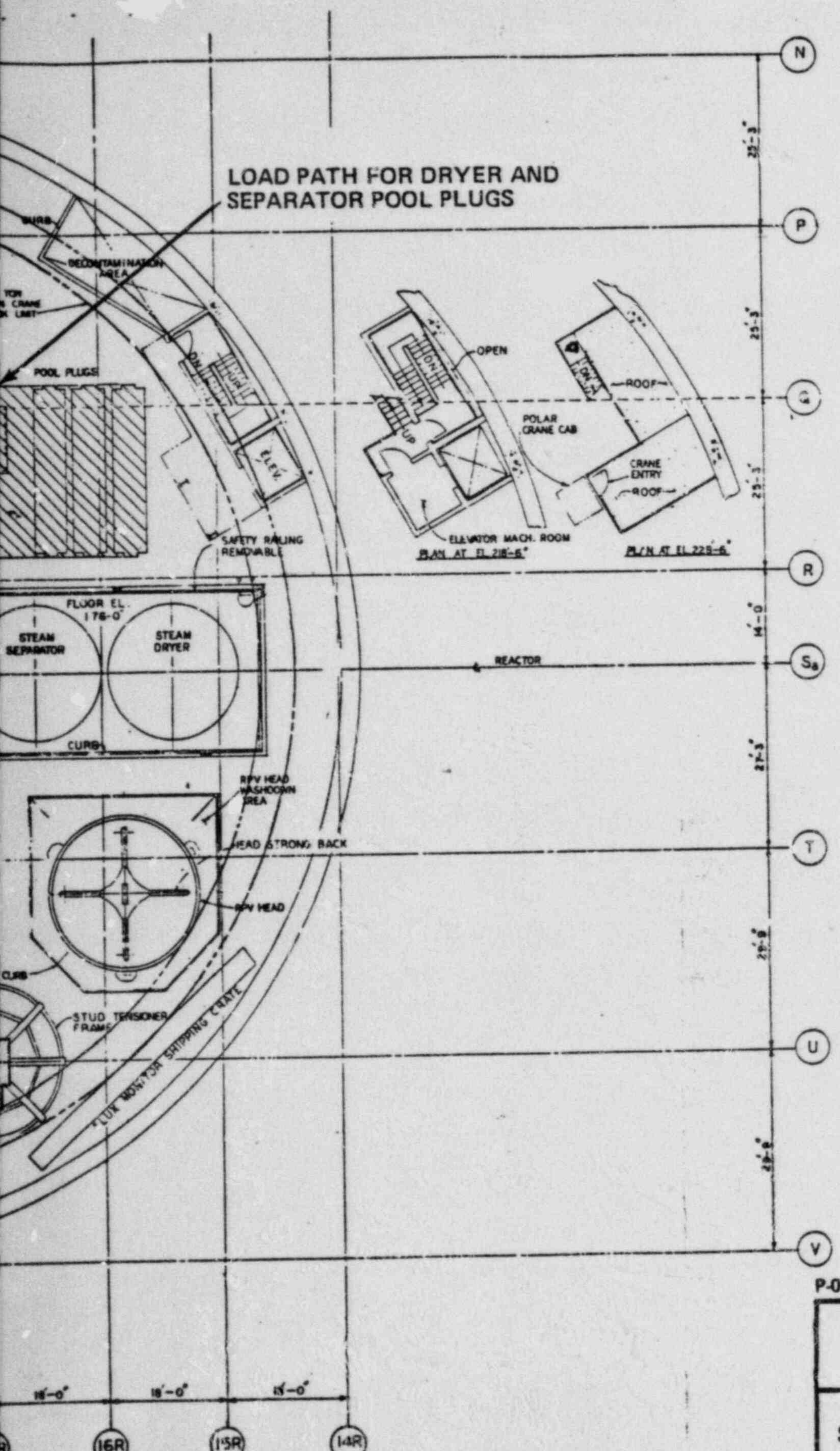
FIGURE 9.1-32
SHEET 3 OF 12

~~AMENDMENT 0, 10/84~~

8409110175-08



LOAD PATH FOR
SPENT FUEL RACK
MODULES AND FUEL
RACK LIFTING FIXTURE
RPV HEAD INSULATION
SHIELD PLUGS (STACKED)
DRYWELL HEAD
POLAR CRANE MAIN HOOK EXCLUSION AREA
POOL PLUG
SKIMMER SURGE TANK (TYP)
SLOT PLUGS (TYP)
JIB CRANE SOCKET (TYP OF 3)
SPENT FUEL POOL
CHANNEL HAND. BOOM SOCKET
EQUIP. HATCH
FUEL PREP. MACH. (TYP)
NEW FUEL VAULT COVERS
CASK LOADING PIT
VESSEL
SAFETY RAILING
SHIELD PLUGS (2)
NEW FUEL INSP. STAND
CASK WASHDOWN AREA
CURB
SACS EXPAN. TANK
SACS EXPANSION TANK
REACTOR
23R
22R
21P
20R
19R
18R



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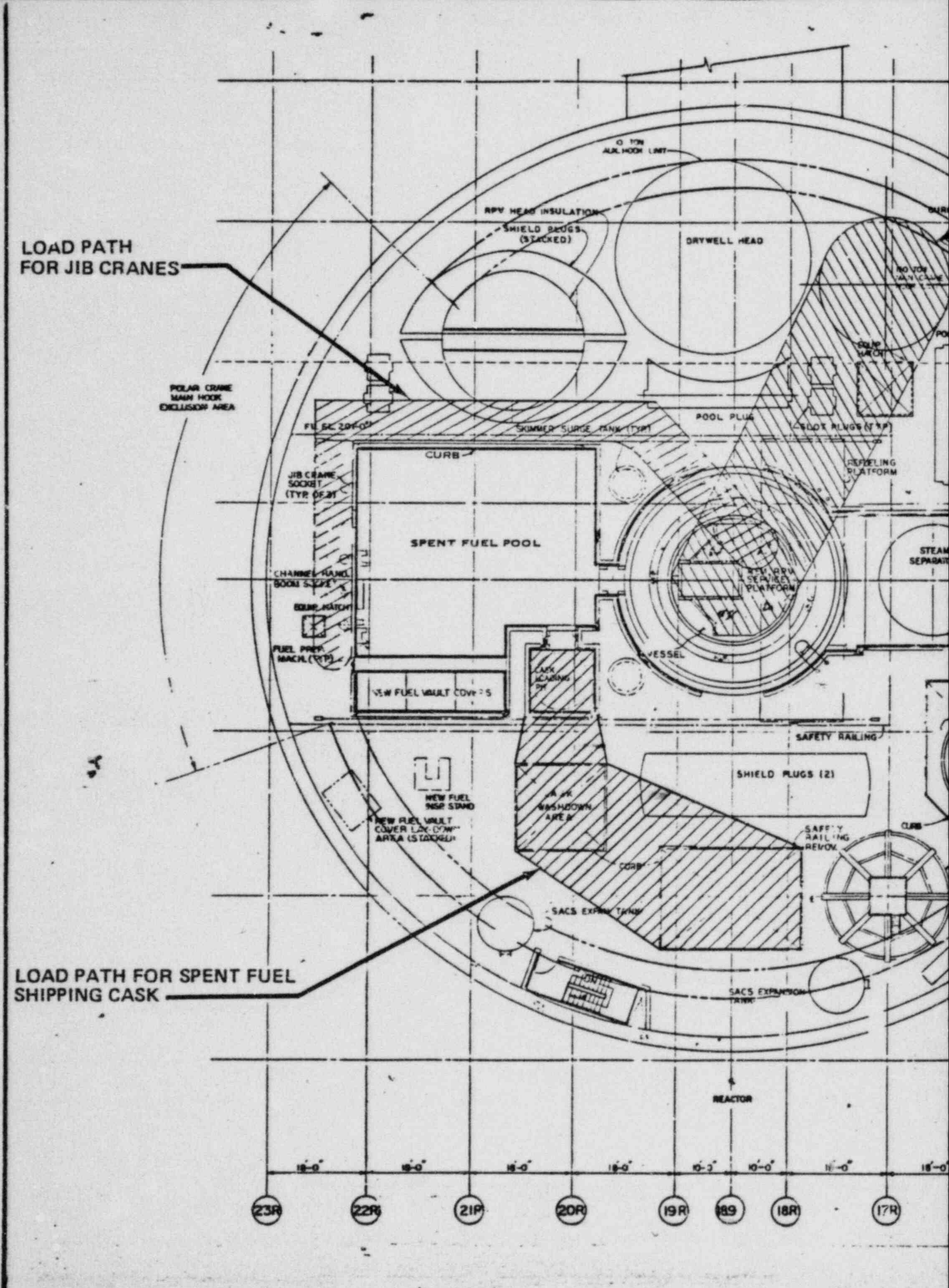
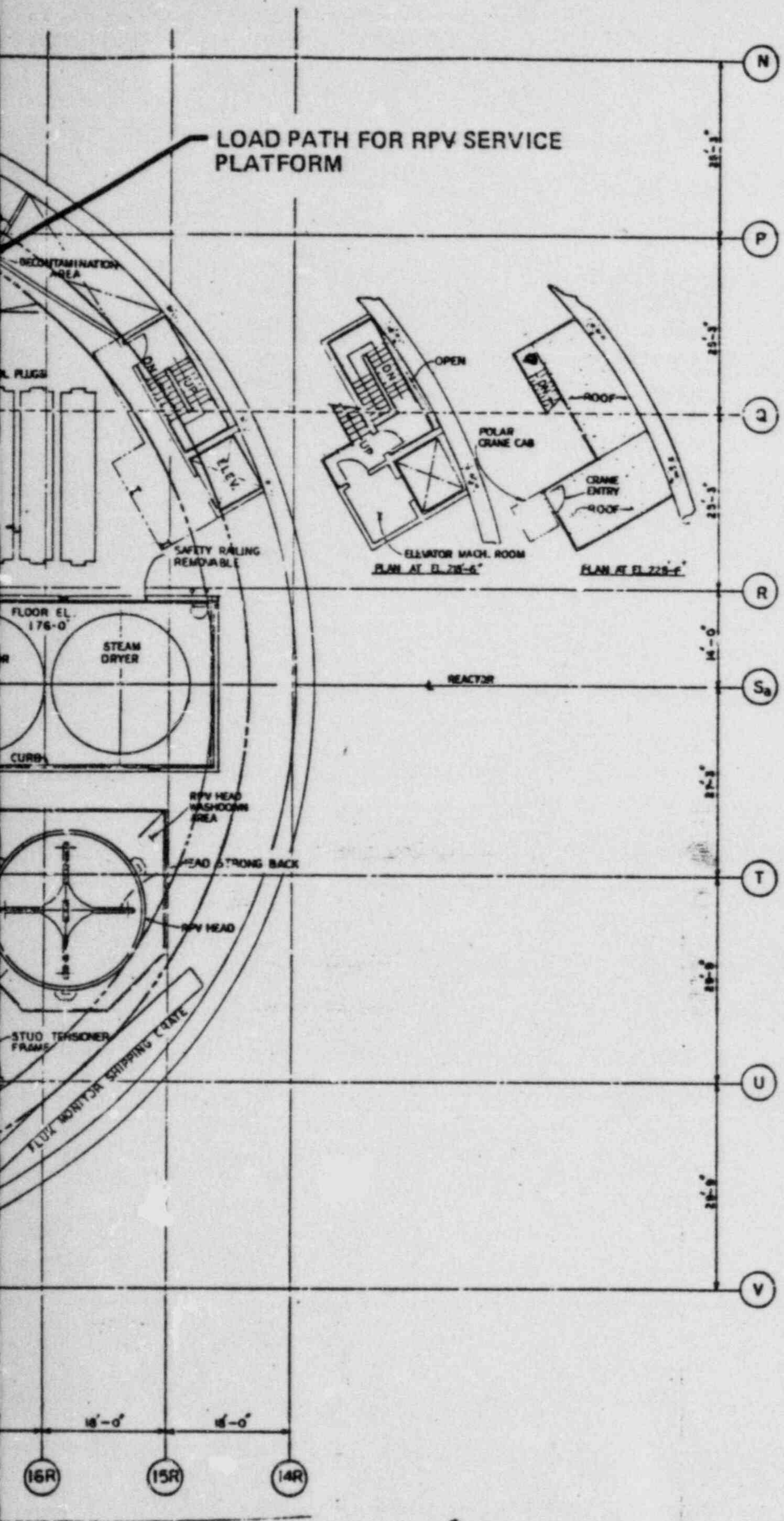
FINAL SAFETY ANALYSIS REPORT

SAFE LOAD PATH DRAWING (EL. 201)
REACTOR BUILDING POLAR CRANE
10H200

FIGURE 9.1-32
SHEET 4 OF 12 ~~AMENDMENT 6, 10/84~~



LOAD PATH
FOR JIB CRANES
LOAD PATH FOR SPENT FUEL
SHIPPING CASK
POLAR CRANE
MAIN HOOK
EXCLUSION AREA
RPV HEAD INSULATION
SHIELD PLUGS
(STACKED)
DRYWELL HEAD
POOL PLUG
SKIMMER SURGE TANK (TYP)
CURB
JIB CRANE
SOCKET
(TYP OF 3)
SPENT FUEL POOL
CHANNEL HANDL
BOOM
EQUIP HATCH
FUEL PREP
MACH. (TYP)
REFUELING
PLATFORM
VESSEL
NEW FUEL VAULT COVERS
CASK
WASHDOWN
AREA
NEW FUEL
INSP STAND
NEW FUEL VAULT
COVER LAY-DOWN
AREA (STAGRD)
SAFETY RAILING
SHIELD PLUGS (2)
SAFETY
RAILING
REMOV.
SACS EXPN TANK
SACS EXPANSION
TANK
REACTOR
18'-0"
10'-0"
11'-0"
23R
22R
21P
20R
19R
189
18R
17R



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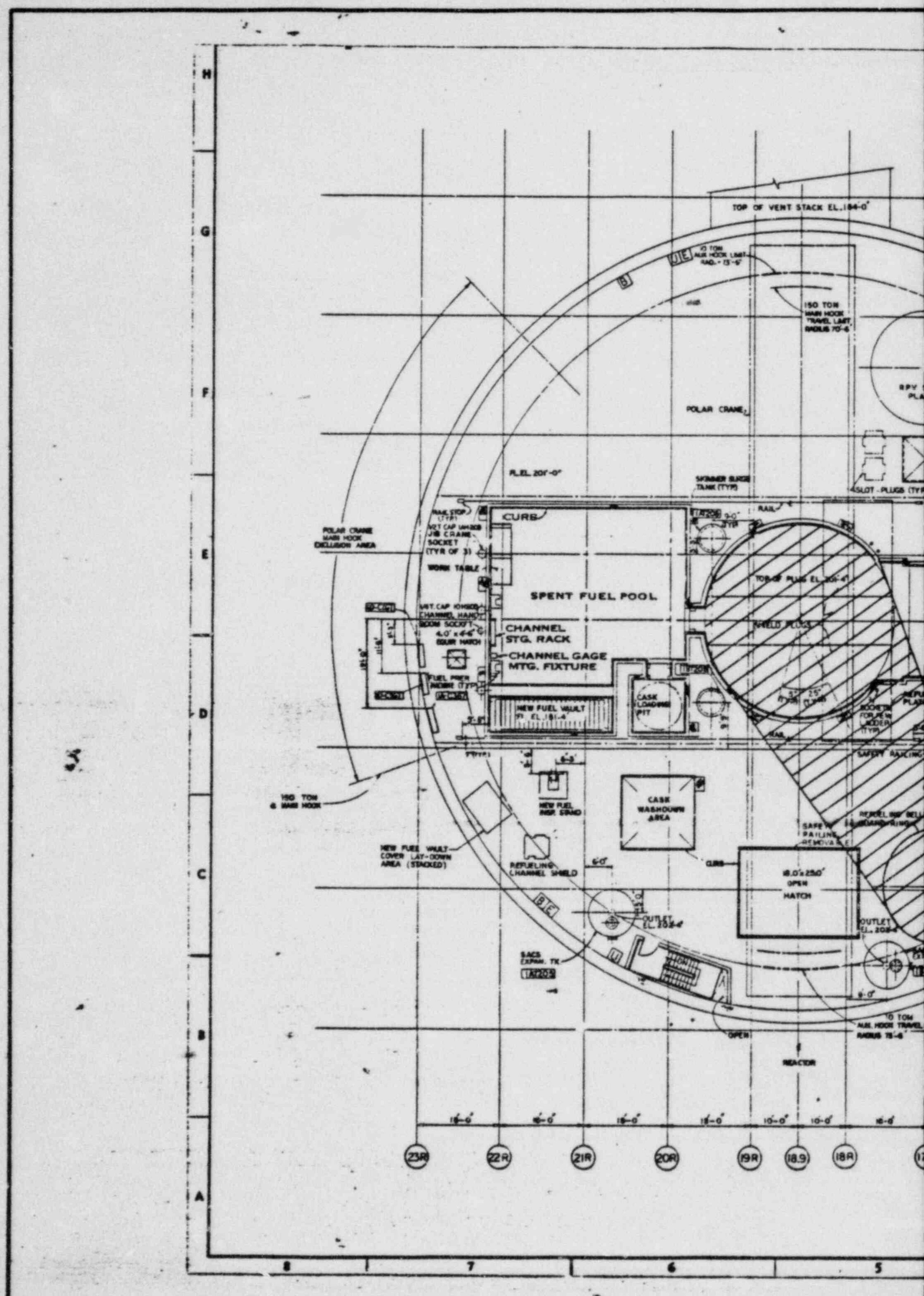
FINAL SAFETY ANALYSIS REPORT

SAFE LOAD PATH DRAWING (EL. 201) REACTOR BUILDING POLAR CRANE 10H200

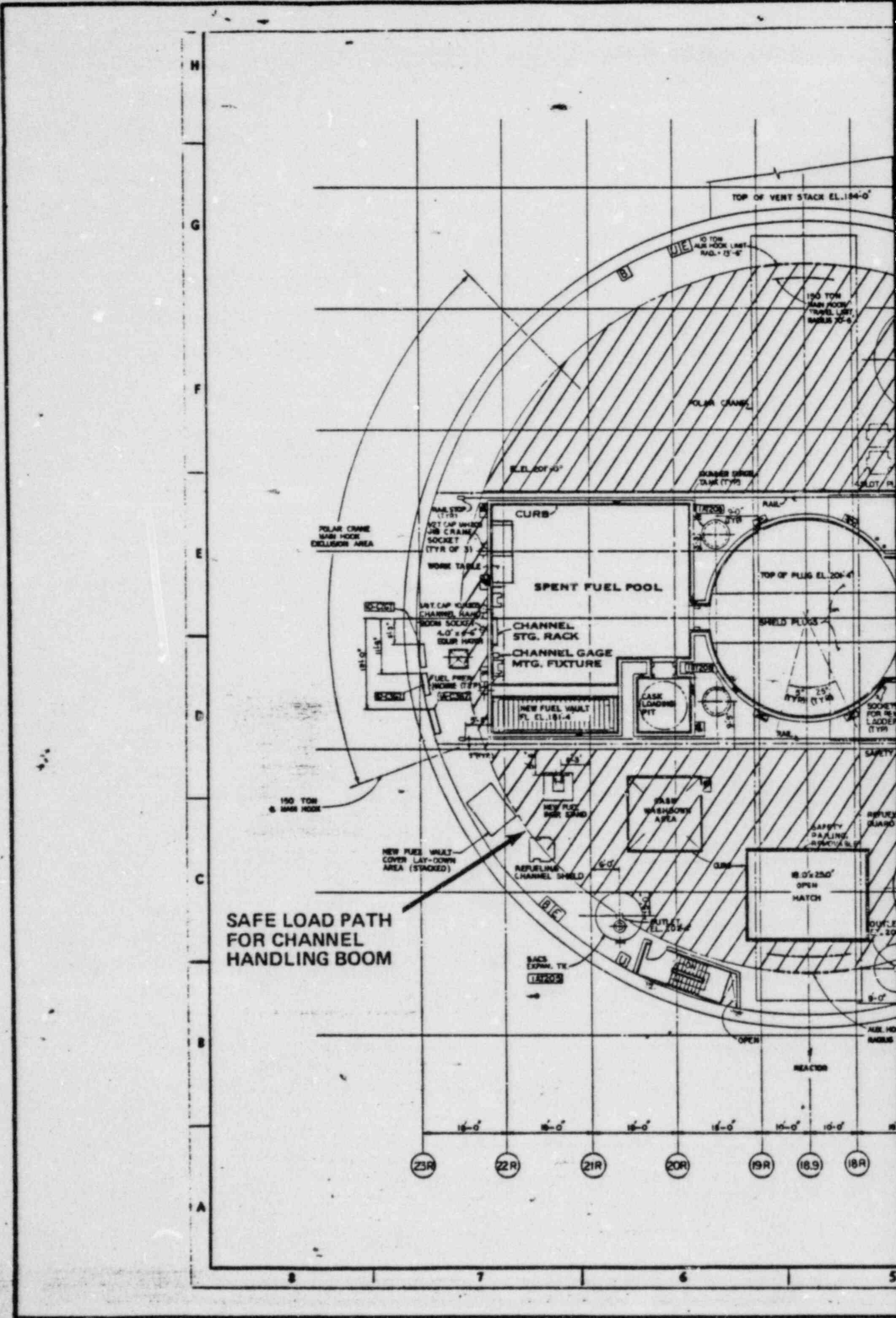
FIGURE 9.1-32
SHEET 5 OF 12

~~AMENDMENT 0, 10/83~~

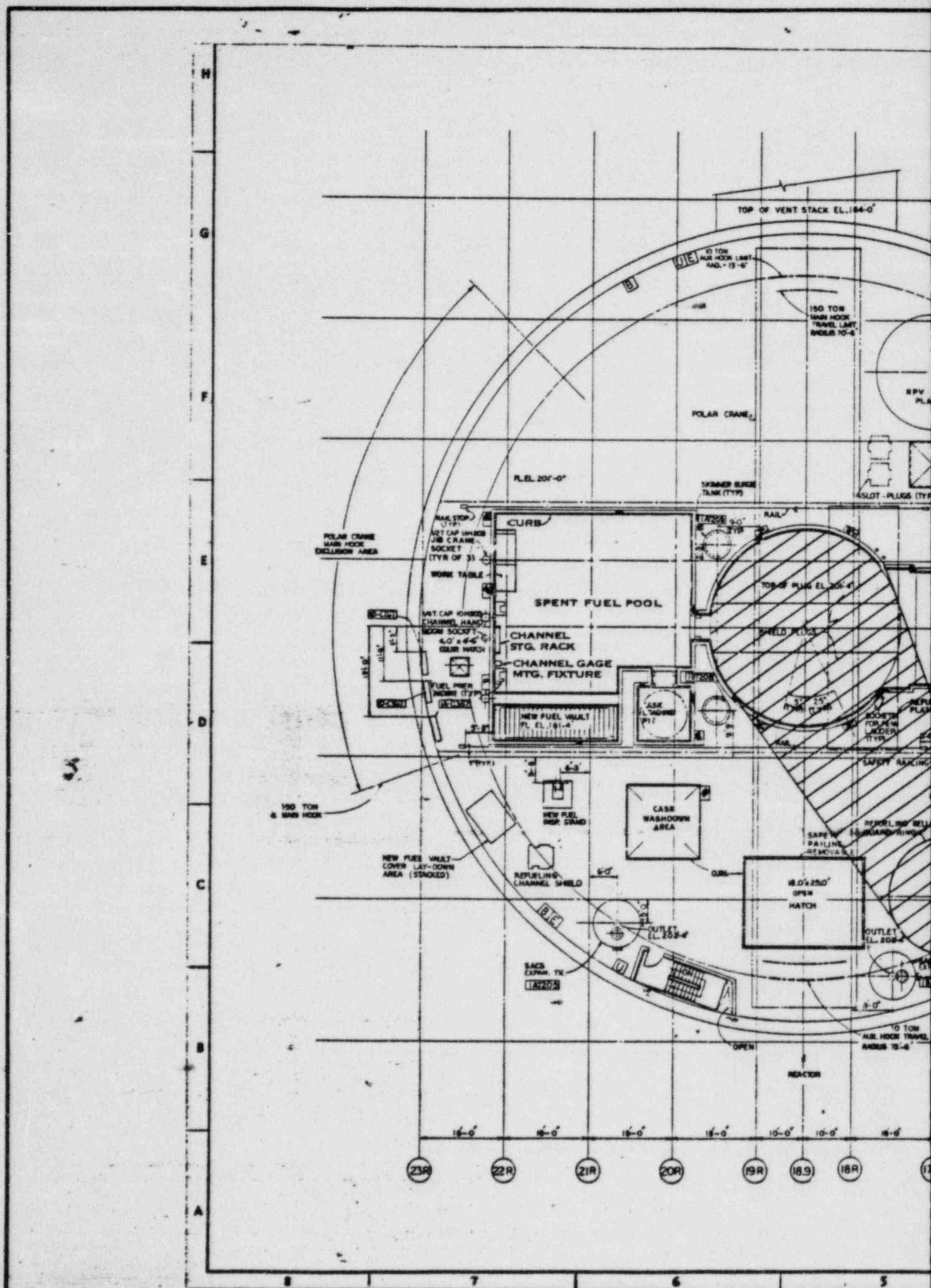
8409110175-/0



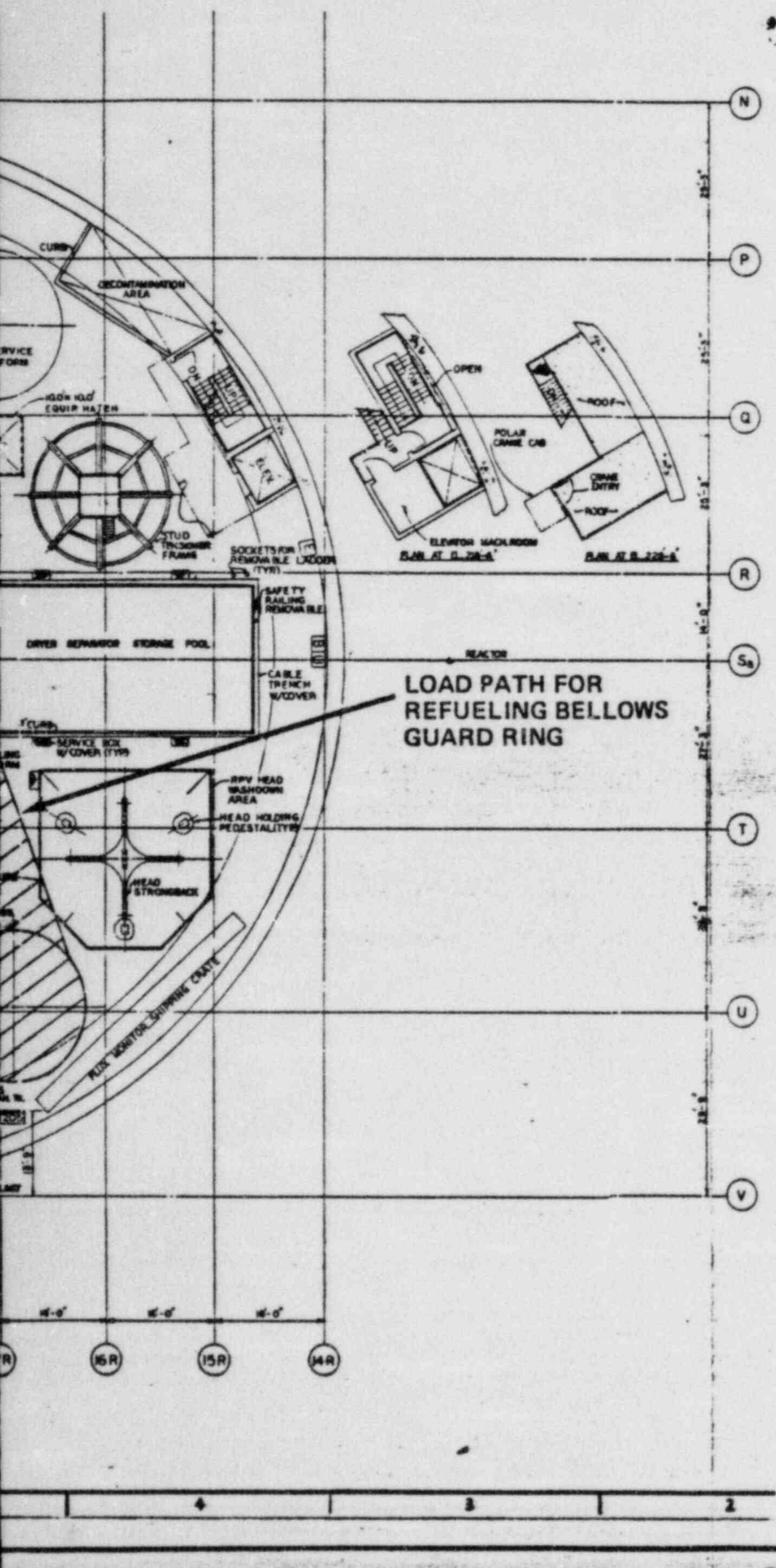
TOP OF VENT STACK EL. 184'-0"
150 TON MAIN HOOK TRAVEL LIMIT, RADIUS 70'-6"
POLAR CRANE
PL. EL. 201'-0"
POLAR CRANE MAIN HOOK EXCLUSION AREA
SKIMMER SURGE TANK (TYP)
SLOT - PLUGS (TYP)
RAIL
CURB
JIB CRANE SOCKET (TYP OF 3)
WORK TABLE
SPENT FUEL POOL
TOP OF PLUG EL. 201'-4"
SHIELD PLUGS
CHANNEL STG. RACK
CHANNEL GAGE MTG. FIXTURE
4.0' x 4'-6" EQUIP. HATCH
NEW FUEL VAULT
CASK LOADING PIT
SAFETY RAILING
150 TON & MAIN HOOK
NEW FUEL INSP. STAND
CASK WASHDOWN AREA
NEW FUEL VAULT COVER LAY-DOWN AREA (STACKED)
REFUELING CHANNEL SHIELD
SAFETY RAILING REMOVABLE
CURB
18.0' x 25.0' OPEN HATCH
OUTLET EL. 208'-6"
SACS EXPAN. TK
OPEN
REACTOR
10 TON AUX. HOOK TRAVEL RADIUS 75'-6"
23R
22R
21R
20R
19R
18.9
18R
H
G
F
E
D
C
B
A
8
7
6
5



SAFE LOAD PATH
FOR CHANNEL
HANDLING BOOM
SPENT FUEL POOL
CURB
CHANNEL
STG. RACK
CHANNEL GAGE
MTG. FIXTURE
NEW FUEL VAULT
CASK LOADING PIT
WORK TABLE
POLAR CRANE MAIN HOOK EXCLUSION AREA
POLAR CRANE
TOP OF VENT STACK
TOP OF PLUG
SHIELD PLUGS
150 TON MAIN HOOK
NEW FUEL VAULT COVER LAY-DOWN AREA (STACKED)
REFUELING CHANNEL SHIELD
OPEN
HATCH
REACTOR
23R
22R
21R
20R
19R
18.9
18R
H
G
F
E
D
C
B
A
8
7
6
5



TOP OF VENT STACK EL. 184-0"
10 TON AUX HOOK LIMIT RAD. = 13'-6"
150 TON MAIN HOOK TRAVEL LIMIT RADIUS 70'-6"
POLAR CRANE
RPV
FL.EL. 201'-0"
SKIMMER SURGE TANK (TYP)
SLOT - PLUGS (TYP)
RAIL
POLAR CRANE MAIN HOOK EXCLUSION AREA
RAIL STOP (TYP)
JIB CRANE SOCKET (TYP OF 3)
WORK TABLE
CURB
SPENT FUEL POOL
TOP OF PLUG EL. 201'-4"
SHIELD PLUGS
CHANNEL STG. RACK
CHANNEL GAGE MTG. FIXTURE
FUEL PREP MACHINE (TYP)
NEW FUEL VAULT FL. EL. 181'-4"
CASK
SAFETY RAILING
150 TON MAIN HOOK
NEW FUEL INSP. STAND
CASK WASHDOWN AREA
NEW FUEL VAULT COVER LAY-DOWN AREA (STACKED)
REFUELING CHANNEL SHIELD
CURB
18.0'x25.0' OPEN HATCH
OUTLET EL. 202'-6"
OUTLET EL. 208'-6"
SAFETY RAILING REMOVABLE
SACS EXPAN. TK
OPEN
10 TON AUX. HOOK TRAVEL RADIUS 78'-6"
REACTOR
16'-0"
16'-0"
16'-0"
16'-0"
16'-0"
10'-0"
10'-0"
16'-0"
23R
22R
21R
20R
19R
18.9
18R
H
G
F
E
D
C
B
A
8
7
6
5



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SAFE LOAD PATH DRAWING (EL. 201)

REACTOR BUILDING POLAR CRANE 10H200

FIGURE 9.1-32
SHEET 6 OF 12

AMENDMENT C, 10/84

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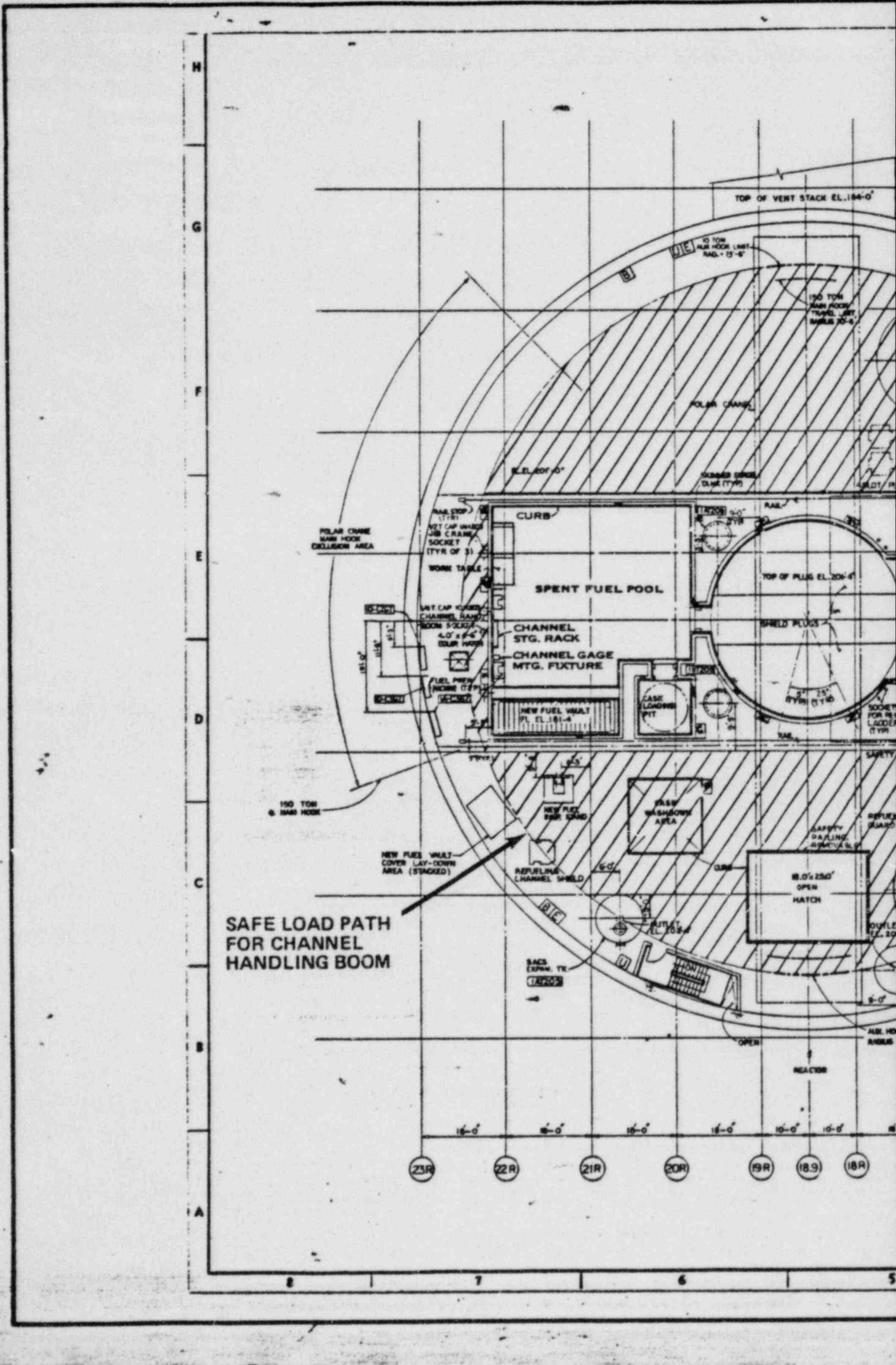
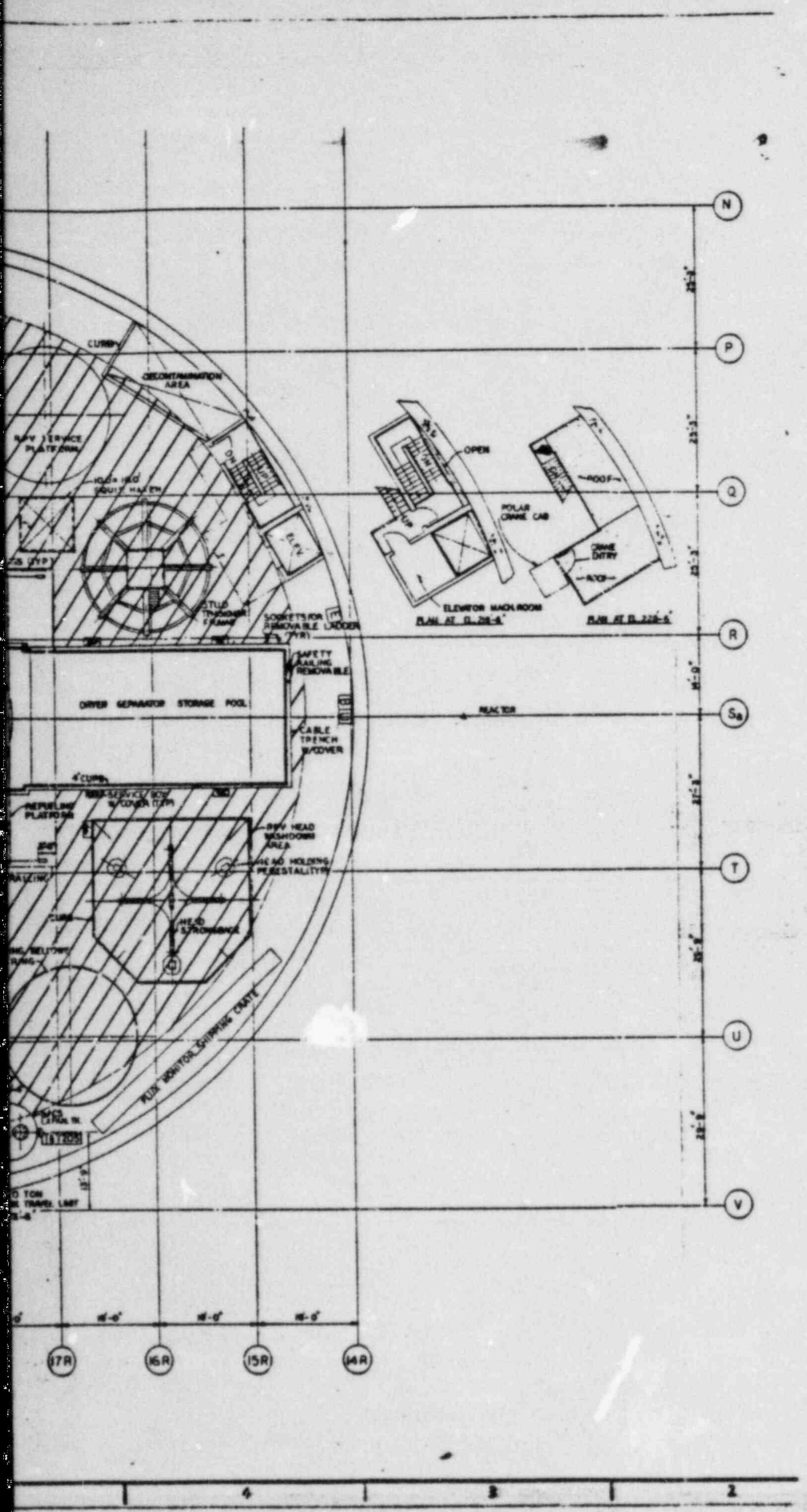


SAFE LOAD PATH
FOR CHANNEL
HANDLING BOOM
SPENT FUEL POOL
CHANNEL
STG. RACK
CHANNEL GAGE
MTG. FIXTURE
NEW FUEL VAULT
FL. EL. 181-4
CURB
TOP OF VENT STACK EL. 184-0"
POLAR CRANE
POLAR CRANE
MAIN HOOK
EXCLUSION AREA
WORK TABLE
SHIELD PLUGS
TOP OF PLUG EL. 201'-4"
CASK
LOADING
PIT
CASK
WASHDOWN
AREA
SAFETY
RAILING
18.0'x25.0'
OPEN
HATCH
NEW FUEL VAULT
COVER LAY-DOWN
AREA (STACKED)
REFUELING
CHANNEL SHIELD
150 TON
MAIN HOOK
OPEN
REACTOR
23R
22R
21R
20R
19R
18.9
18R



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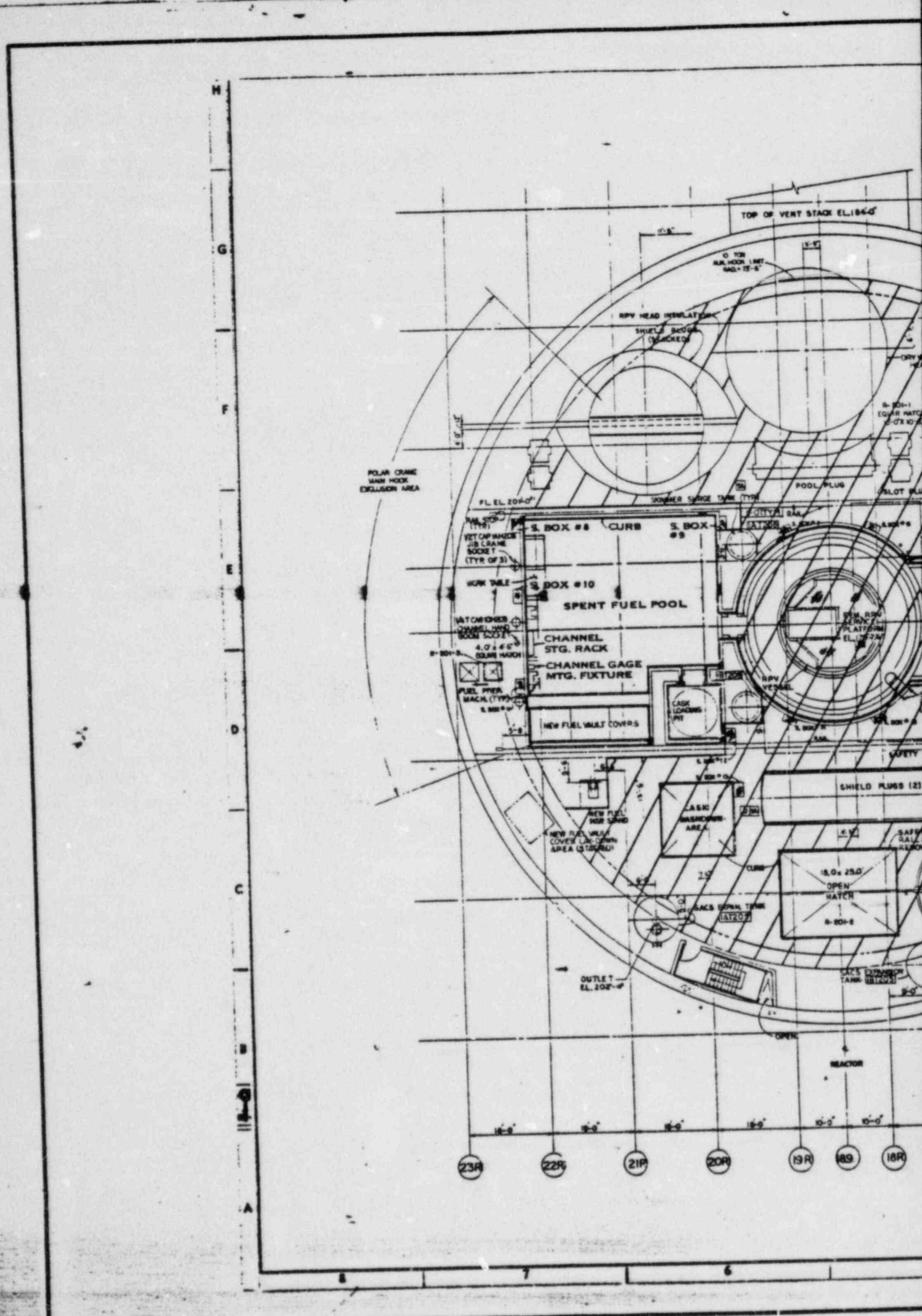
FINAL SAFETY ANALYSIS REPORT

SAFE LOAD PATH DRAWING (EL. 201
REACTOR BUILDING POLAR CRANE
10H200

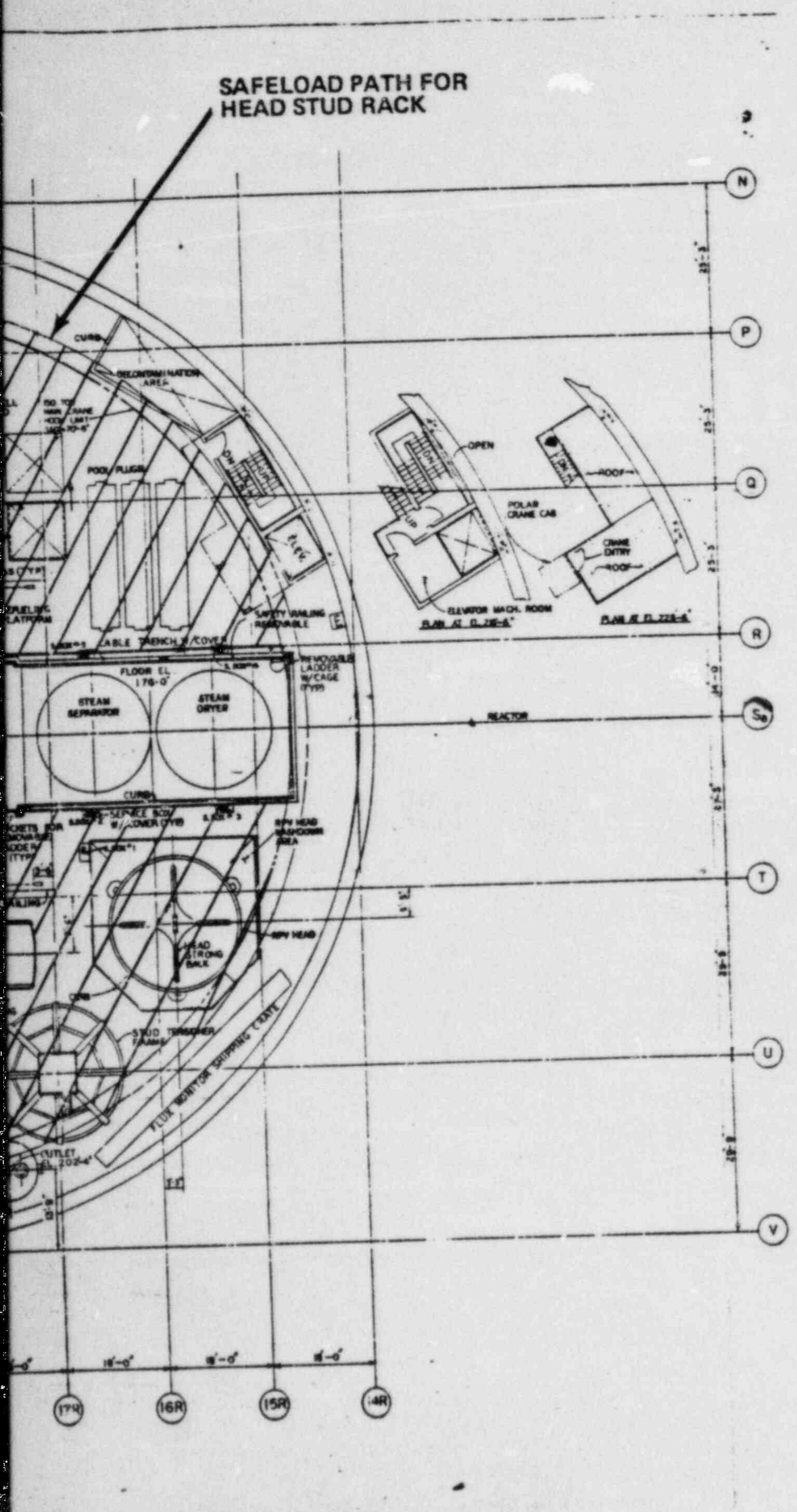
FIGURE 9.1-32
SHEET 7 OF 12

8409110175-12



H
G
F
E
D
C
B
A
TOP OF VENT STACK EL.186'-0"
RPV HEAD INSULATION
SHIELD BLOCKS (STACKED)
POLAR CRANE MAIN HOOK EXCLUSION AREA
FL. EL. 201'-0"
SKIMMER SURGE TANK (TYP)
POOL PLUG
S. BOX #8
CURB
S. BOX #9
WORK TABLE
S. BOX #10
SPENT FUEL POOL
CHANNEL STG. RACK
CHANNEL GAGE MTG. FIXTURE
4.0' x 4.5' SQUARE HATCH
NEW FUEL VAULT COVERS
CASK LOADING PIT
RPV VESSEL
SHIELD PLUGS (2)
CASK WASHDOWN AREA
NEW FUEL VAULT COVER LAY DOWN AREA (STACKED)
CURB
18.0' x 25.0' OPEN HATCH
SACS EXPANSION TANK
OUTLET EL. 202'-4"
OPEN
REACTOR
18'-0"
18'-0"
10'-0"
10'-0"
23R
22R
21R
20R
19R
18.9
18R
8
7
6

SAFELOAD PATH FOR HEAD STUD RACK



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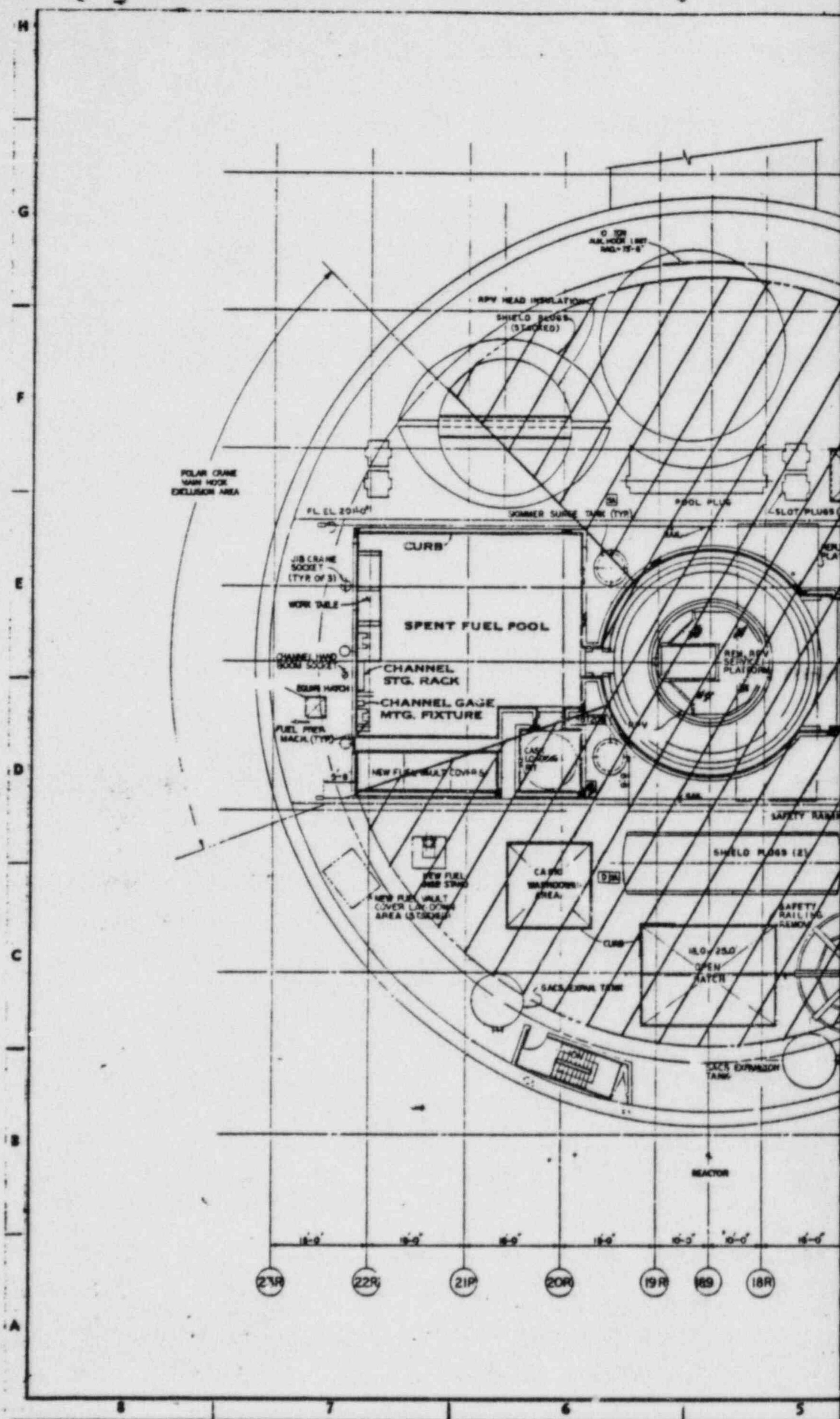
SAFE LOAD PATH DRAWING (EL. 201)

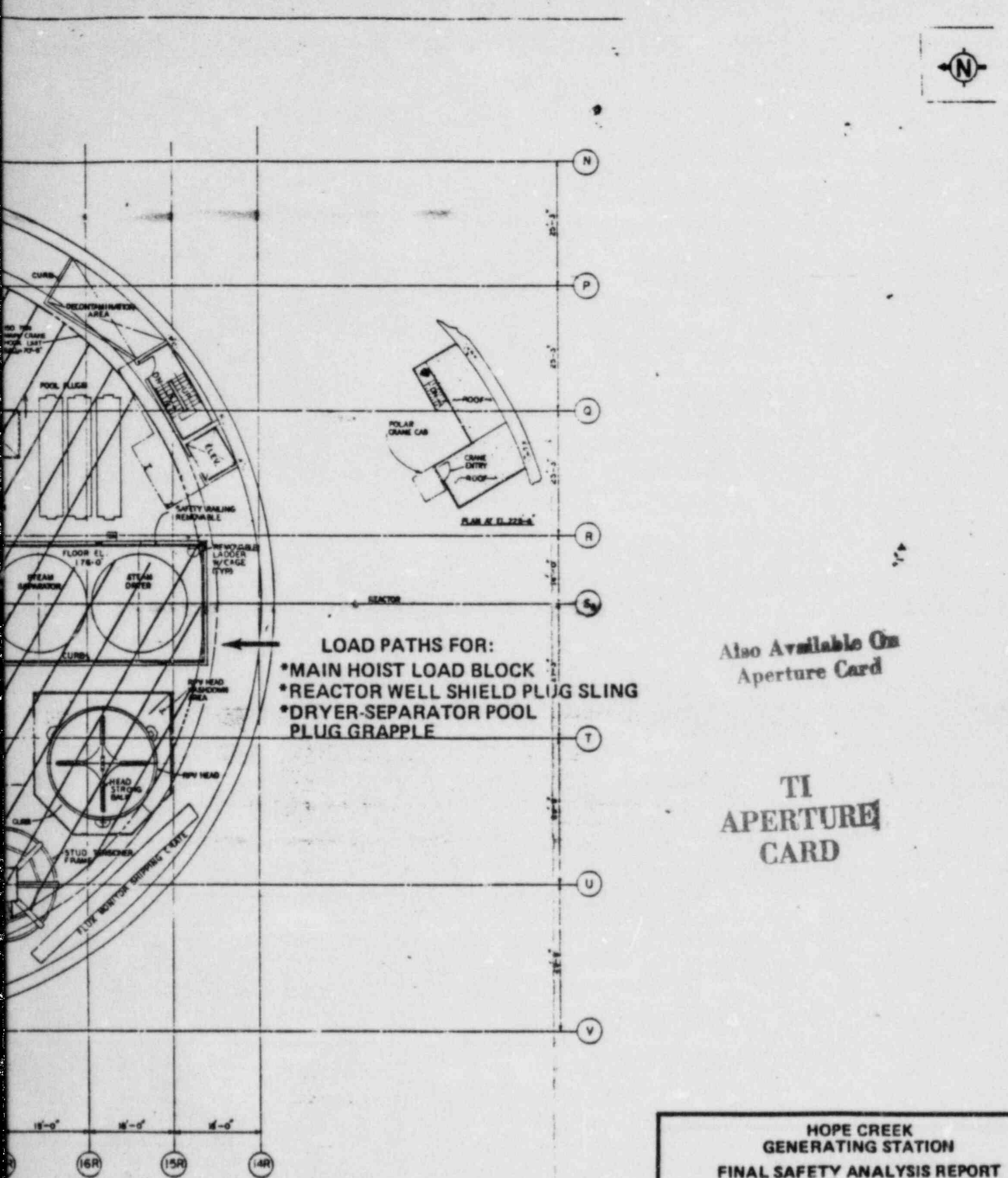
REACTOR BUILDING POLAR CRANE 10H200

FIGURE 9.1-32 SHEET 8 OF 12

AMENDMENT 8, 10/8

8409110175-/3


SPENT FUEL POOL
CURB
CHANNEL STG. RACK
CHANNEL GAGE MTG. FIXTURE
WORK TABLE
JIB CRANE SOCKET (TYP OF 3)
POLAR CRANE MAIN HOOK EXCLUSION AREA
RPV HEAD INSULATION
SHIELD PLUGS (STACKED)
POOL PLUG
SHIELD PLUGS (2)
CASK WASHDOWN AREA
OPEN HATCH
CURB
REACTOR
23R
22R
21R
20R
19R
18R
H
G
F
E
D
C
B
A
8
7
6
5



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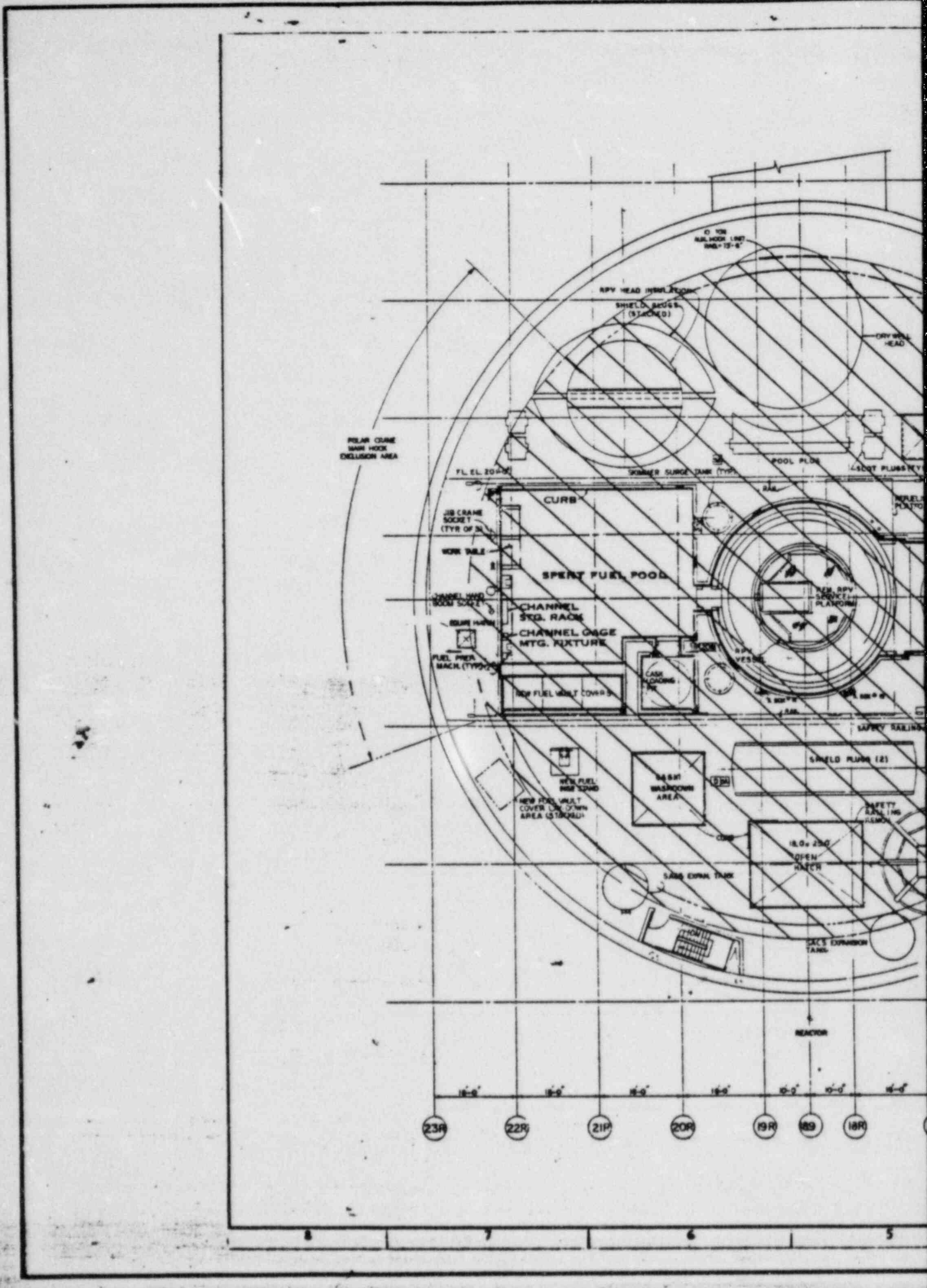
SAFE LOAD PATH DRAWING (EL. 201)

REACTOR BUILDING POLAR CRANE 10H200

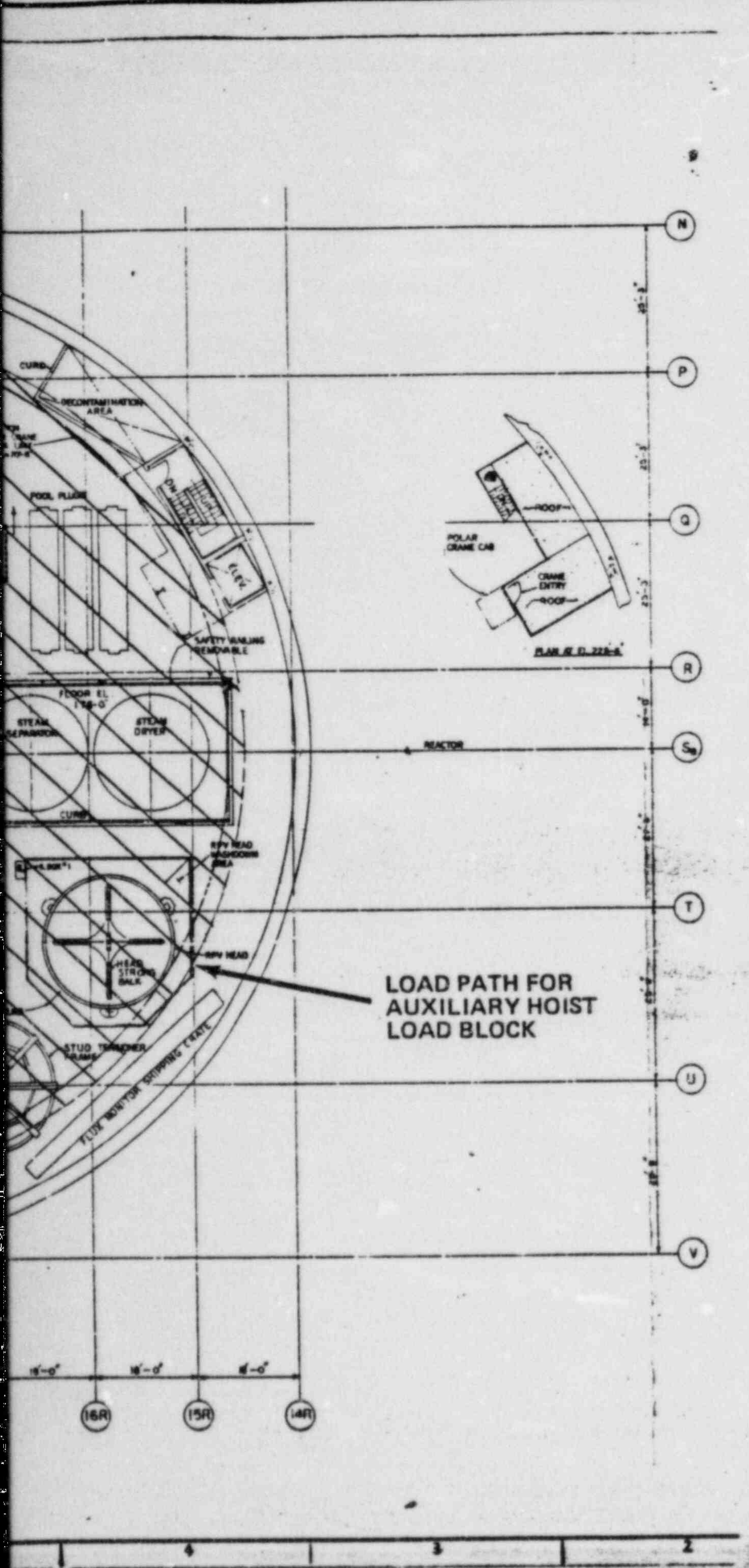
FIGURE 9.1-32
~~SHEET 9 OF 12~~

~~AMENDMENT 6, 10/84~~

8409110175-14



POLAR CRANE MAIN HOOK EXCLUSION AREA
RPV HEAD INSULATION
SHIELD PLUGS (STACKED)
DRY WELL HEAD
POOL PLUG
CURB
JIB CRANE SOCKET (TYP. OF 3)
WORK TABLE
SPENT FUEL POOL
CHANNEL STG. RACK
CHANNEL GAGE MTG. FIXTURE
FUEL PREP. MACH. (TYP.)
NEW FUEL VAULT COVERS
CASK LOADING PIT
RPV VESSEL
SAFETY RAILING
SHIELD PLUGS (2)
WASHDOWN AREA
NEW FUEL VAULT COVER LAY DOWN AREA (STAGING)
OPEN HATCH
SACS EXPANSION TANK
REACTOR
23R
22R
21R
20R
19R
189
18R
8
7
6
5



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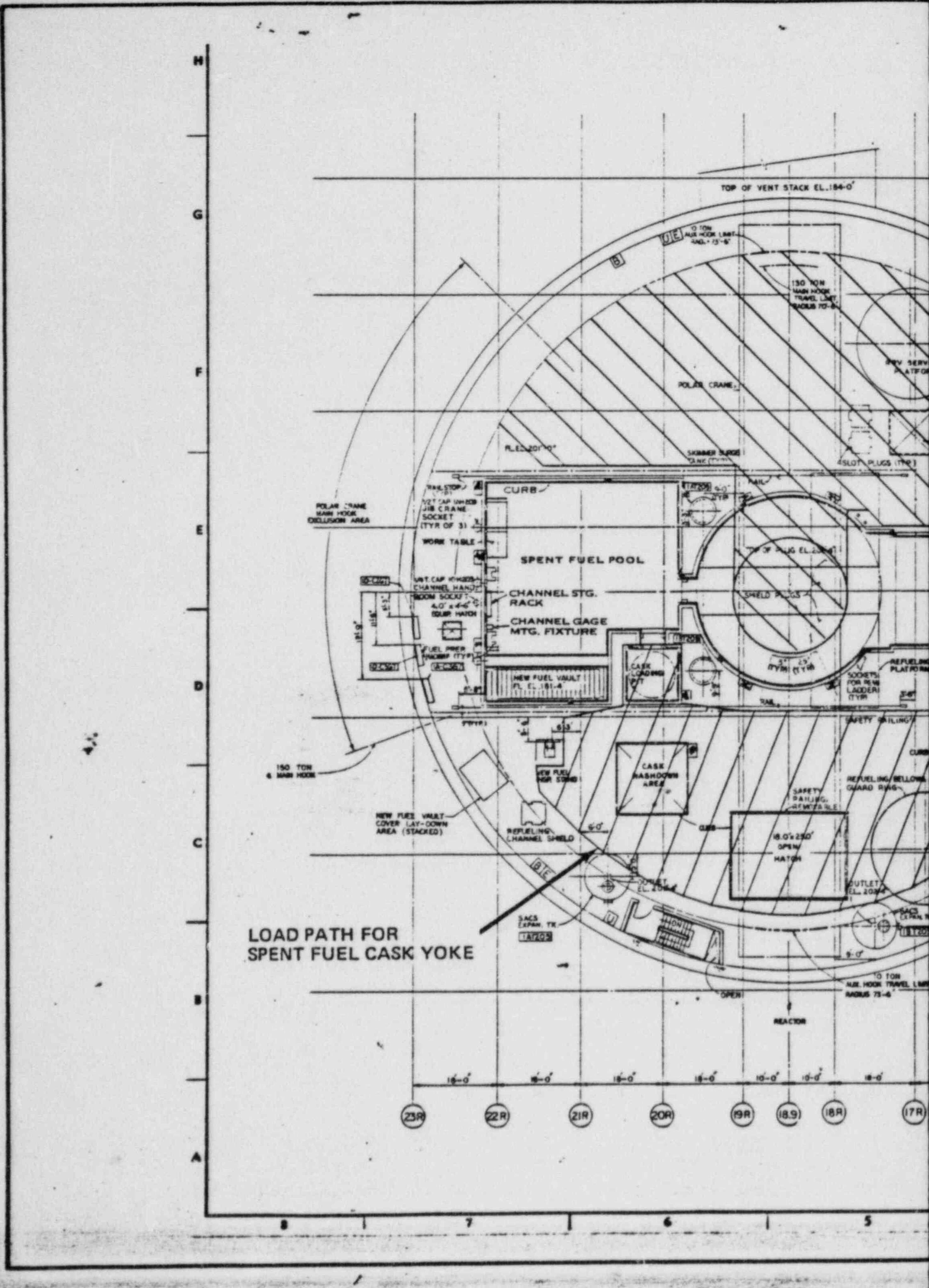
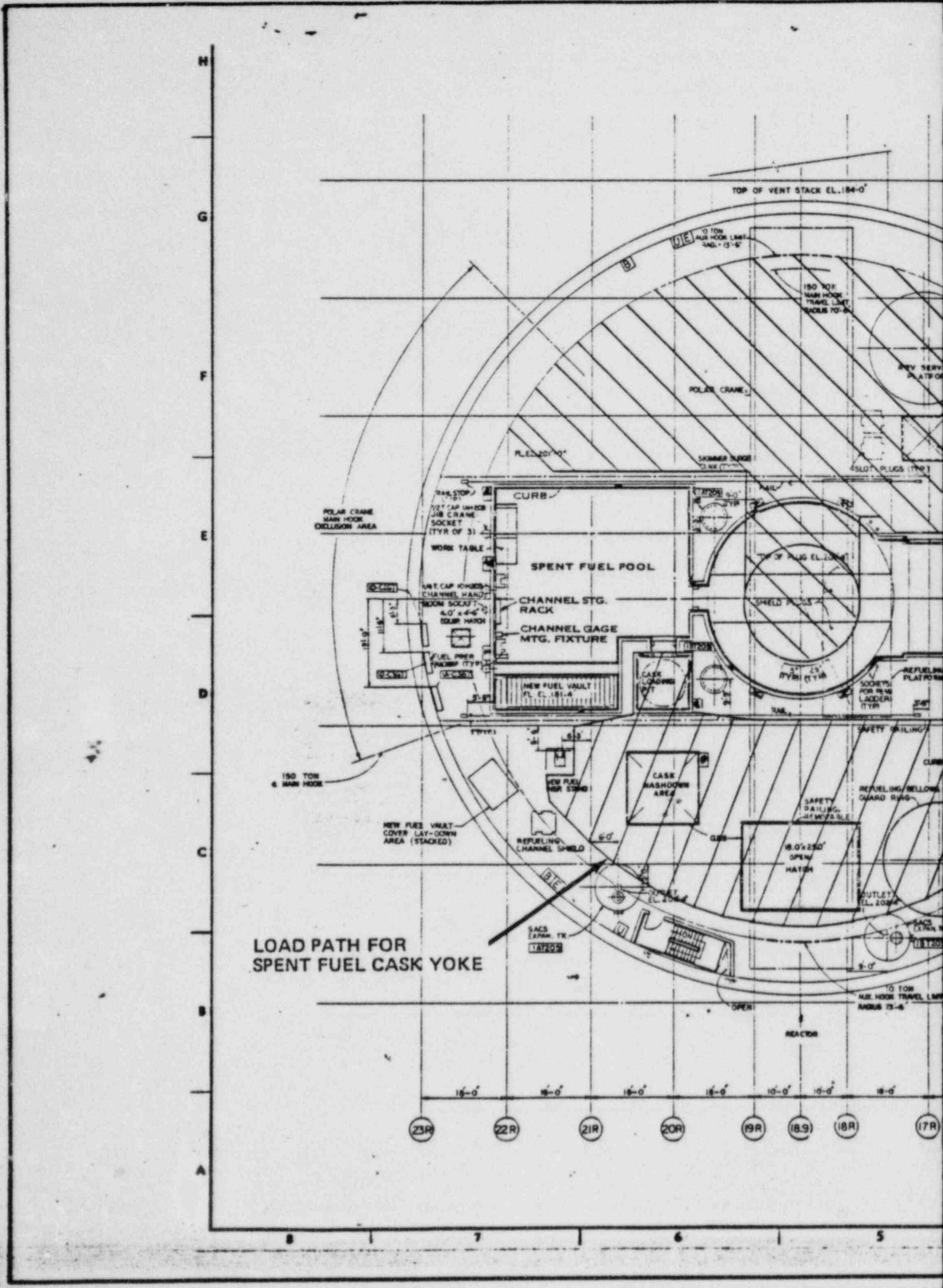
FINAL SAFETY ANALYSIS REPORT

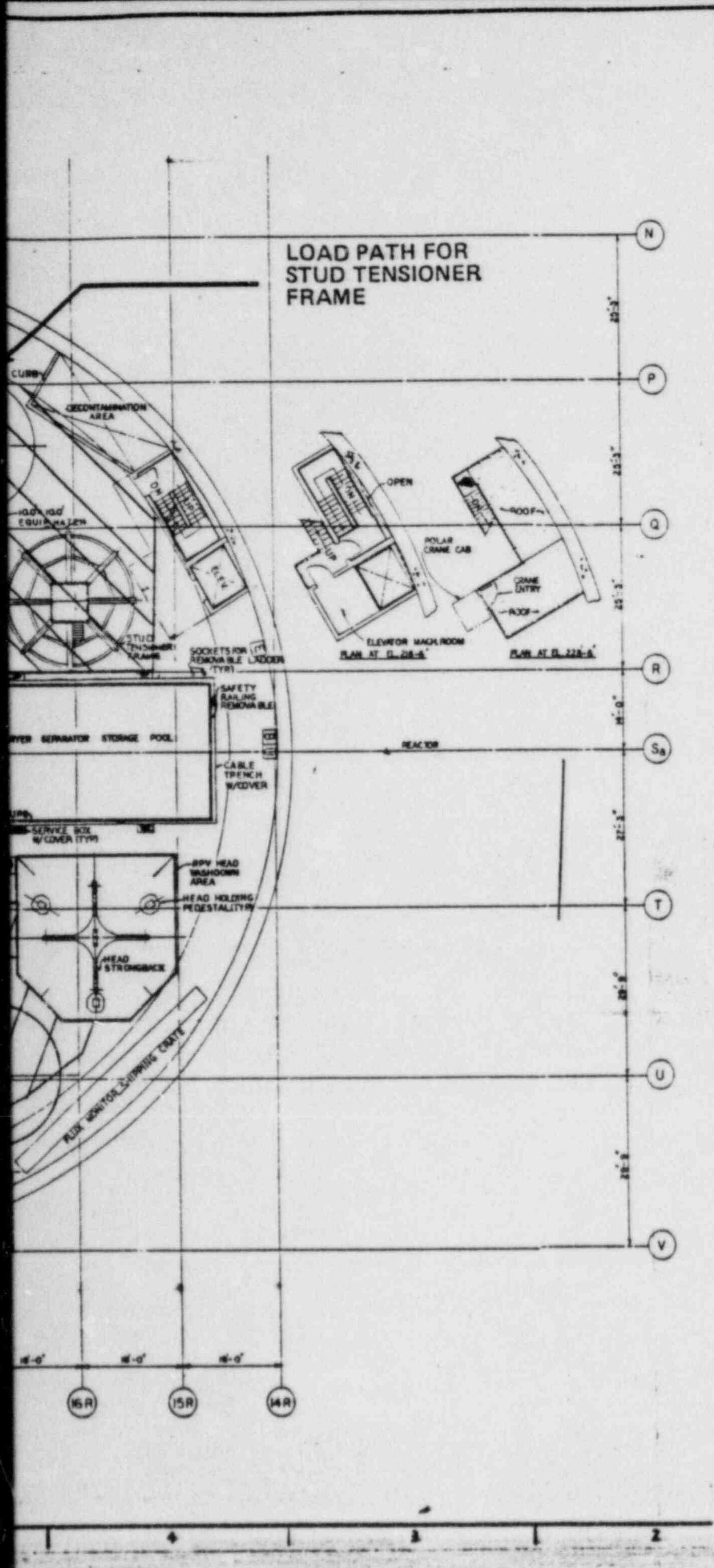
SAFE LOAD PATH DRAWING (EL. 201)
REACTOR BUILDING POLAR CRANE
10H200

FIGURE 9.1-32
~~SHEET 10 OF 12~~ ~~AMENDMENT 6, 10/84~~



TOP OF VENT STACK EL.184-0"
POLAR CRANE
CURB
WORK TABLE
SPENT FUEL POOL
CHANNEL STG. RACK
CHANNEL GAGE MTG. FIXTURE
NEW FUEL VAULT
CASK WASHDOWN AREA
18.0'x25.0' OPEN HATCH
REFUELING CHANNEL SHIELD
150 TON ℄ MAIN HOOK
NEW FUEL VAULT COVER LAY-DOWN AREA (STACKED)
LOAD PATH FOR SPENT FUEL CASK YOKE
REACTOR
23R
22R
21R
20R
19R
18.9
18R
17R
H
G
F
E
D
C
B
A
8
7
6
5


TOP OF VENT STACK EL. 184-0"
POLAR CRANE
SPENT FUEL POOL
CHANNEL STG. RACK
CHANNEL GAGE MTG. FIXTURE
NEW FUEL VAULT
CASK LOADING PIT
CASK WASHDOWN AREA
18.0'x25.0' OPEN HATCH
SHIELD PLUGS
SAFETY RAILING
REFUELING BELLOWS GUARD RING
POLAR CRANE MAIN HOOK EXCLUSION AREA
WORK TABLE
CURB
150 TON MAIN HOOK
NEW FUEL VAULT COVER LAY-DOWN AREA (STACKED)
REFUELING CHANNEL SHIELD
LOAD PATH FOR SPENT FUEL CASK YOKE
REACTOR
23R
22R
21R
20R
19R
18.9
18R
17R
H
G
F
E
D
C
B
A
8
7
6
5



P-0047-1, REV 7

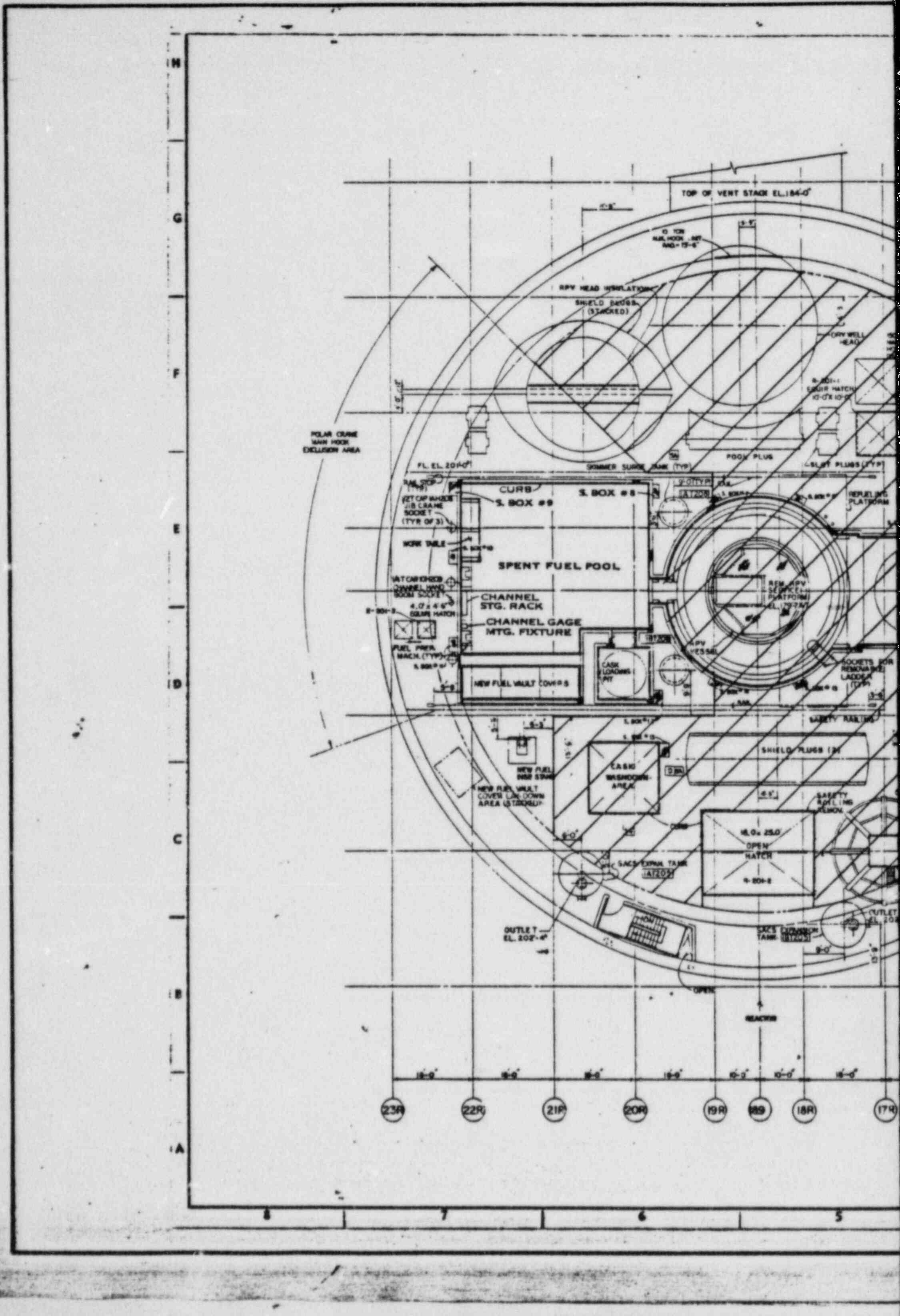
HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

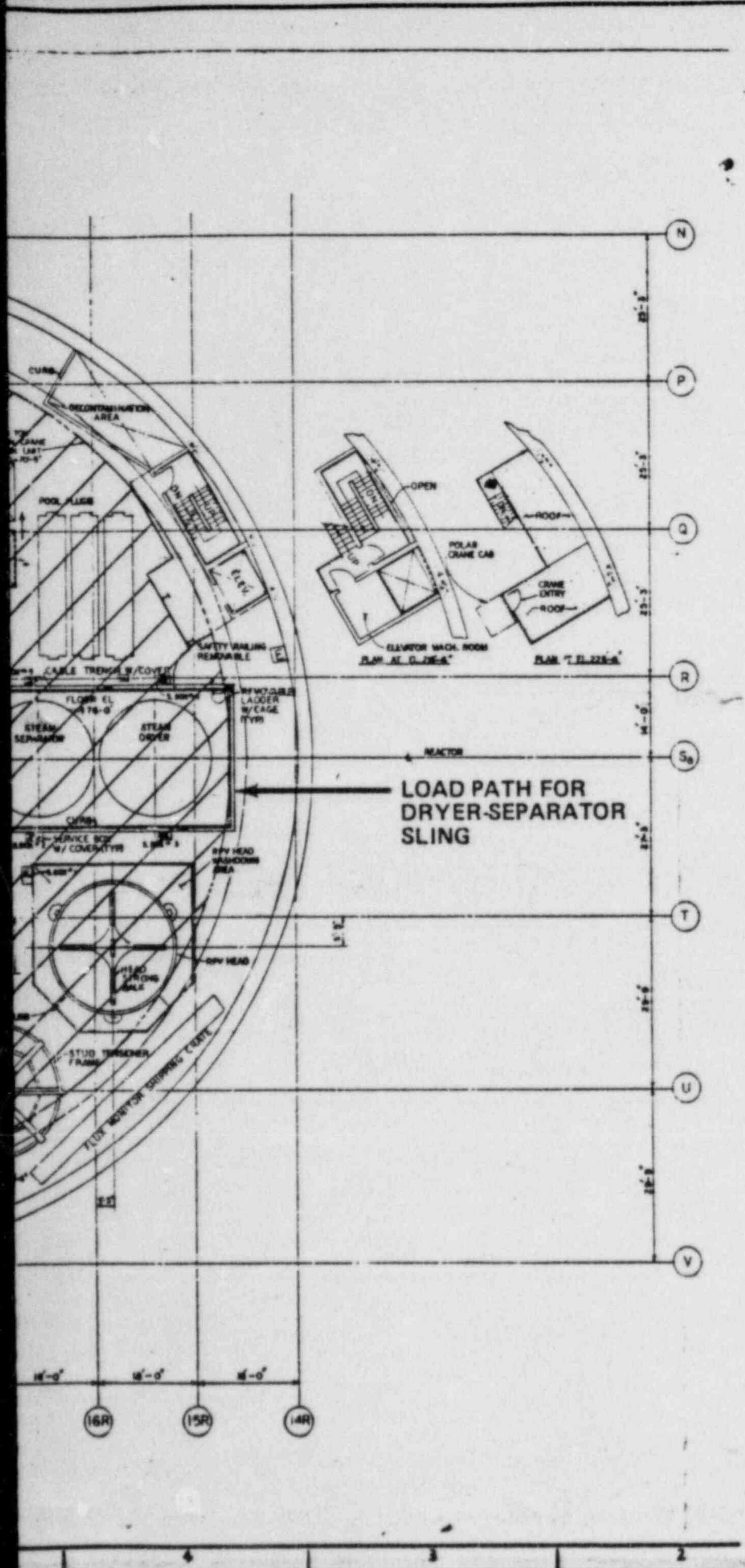
SAFE LOAD PATH DRAWING (EL. 201)

REACTOR BUILDING POLAR CRANE
10H200

FIGURE 9.1-32
SHEET 11 OF 12

~~AMENDMENT 0, 10/84~~


TOP OF VENT STACK EL. 184'-0"
RPV HEAD INSULATION
SHIELD PLUGS (STACKED)
DRY WELL HEAD
POLAR CRANE MAIN HOOK EXCLUSION AREA
POOL PLUG
SLOT PLUGS (TYP)
SKIMMER SURGE TANK (TYP)
REFUELING PLATFORM
CURB
S. BOX #9
S. BOX #8
SPENT FUEL POOL
WORK TABLE
CHANNEL STG. RACK
CHANNEL GAGE MTG. FIXTURE
FUEL PREP. MACH. (TYP)
NEW FUEL VAULT COVERS
CASK LOADING PIT
RPV VESSEL
SAFETY RAILING
SHIELD PLUGS (3)
CASK WASHDOWN AREA
NEW FUEL VAULT COVER LAY DOWN AREA (STORAGE)
18.0 x 25.0 OPEN HATCH
SACS EXPAN. TANK
OUTLET EL. 202'-4"
OPEN
REACTOR
23R
22R
21R
20R
19R
18R
17R
H
G
F
E
D
C
B
A
8
7
6
5



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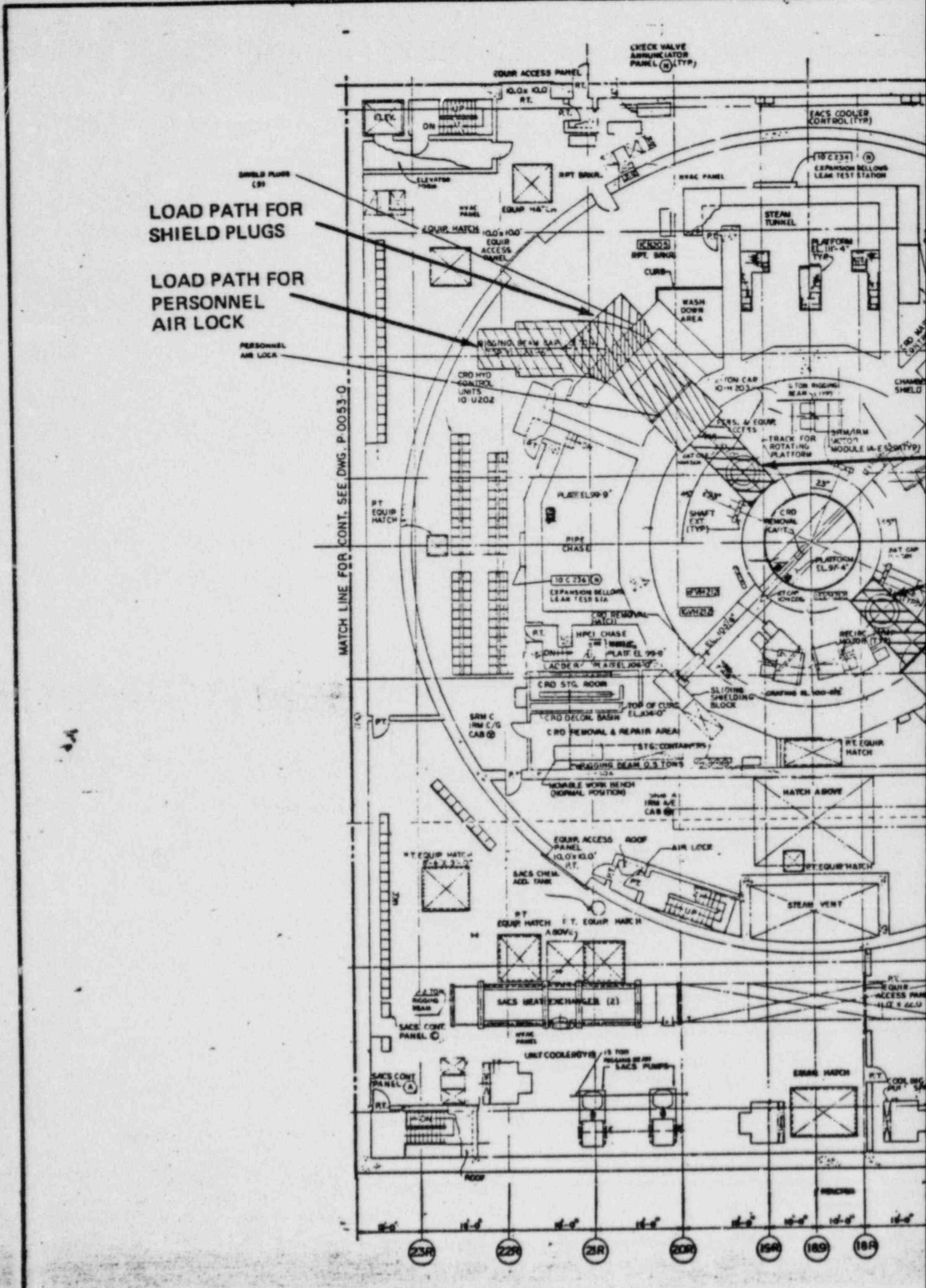
HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

SAFE LOAD PATH DRAWING (EL. 201)
REACTOR BUILDING POLAR CRANE
10H200

FIGURE 9.1-32
SHEET 12 OF 12

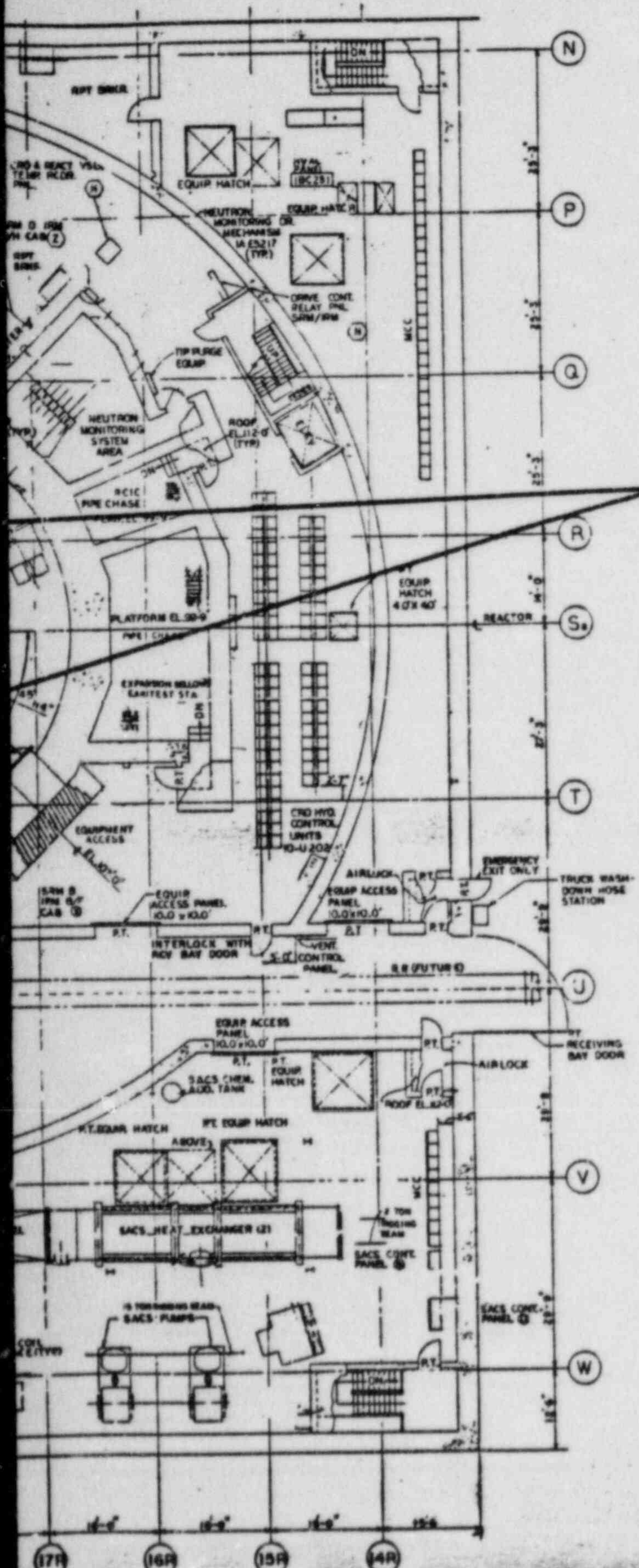
8409110175-/7


LOAD PATH FOR
SHIELD PLUGS
LOAD PATH FOR
PERSONNEL
AIR LOCK
SHIELD PLUGS
PERSONNEL
AIR LOCK
CHECK VALVE
ANNUNCIATOR
PANEL
EQUIP. ACCESS PANEL
ELEV.
RPT. BRKR.
HVAC PANEL
EQUIP. HATCH
STEAM
TUNNEL
WASH
DOWN
AREA
CRD HYD
CONTROL
UNITS
10-U202
TRACK FOR
ROTATING
PLATFORM
SHAFT
EXT
PIPE
CHASE
CRD
REMOVAL
EXPANSION BELLOWS
LEAK TEST STA
CRD REMOVAL
HATCH
HPCI CHASE
CRD STG. ROOM
CRD DECON. BASIN
CRD REMOVAL & REPAIR AREA
SLIDING
SHIELDING
BLOCK
MOVABLE WORK BENCH
(NORMAL POSITION)
HATCH ABOVE
AIR LOCK
STEAM VENT
SACS CHEM.
ADD. TANK
SACS HEAT EXCHANGER (2)
SACS CONT.
PANEL
UNIT COOLERS
SACS PUMPS
SACS CONT
PANEL
MATCH LINE FOR CONT. SEE DWG. P-0053-0
23R
22R
21R
20R
19R
18.9
18R

NOTE:

THE LOAD PATHS ON THIS DRAWING ARE FOR:

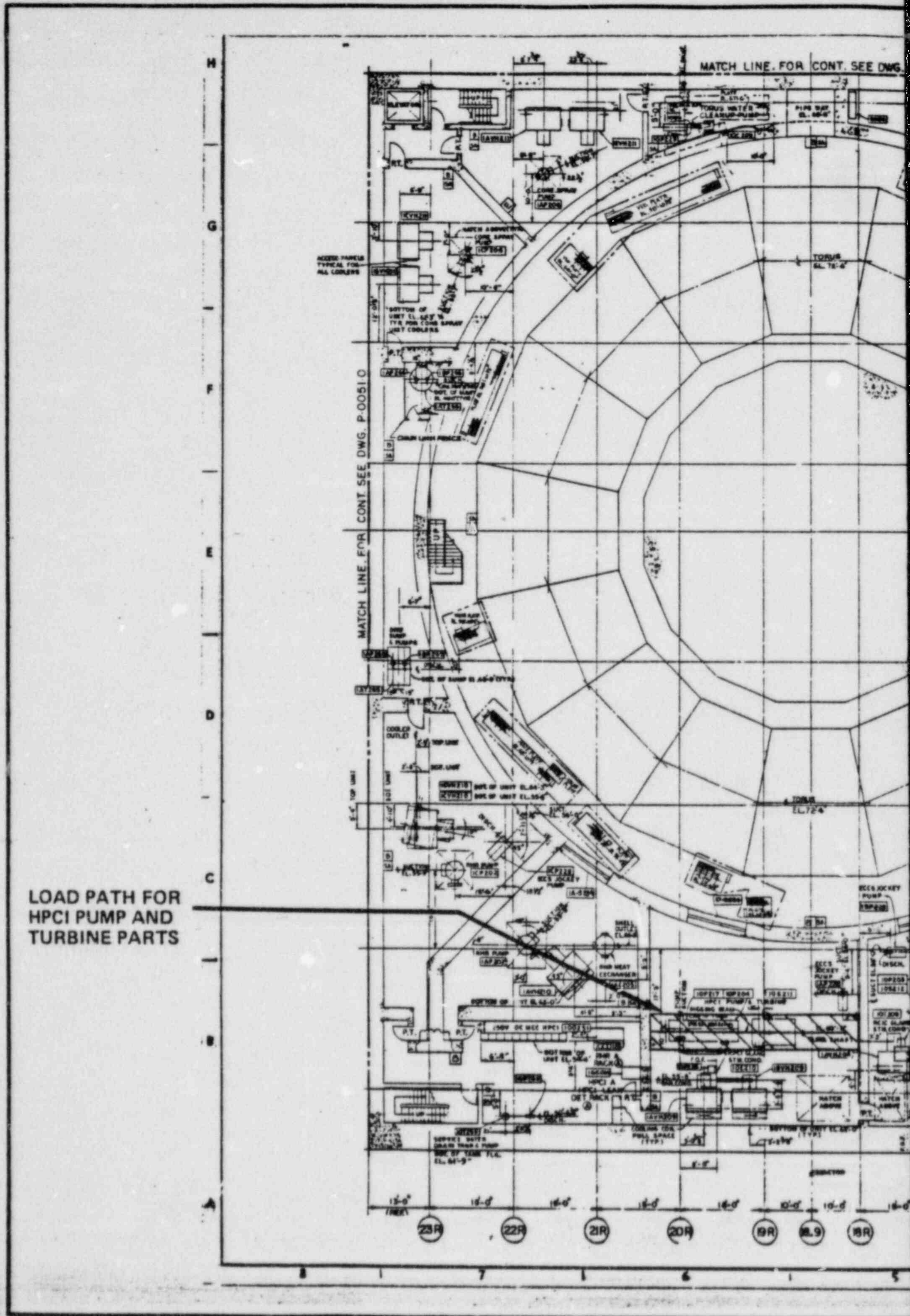
1OH217	PERSONNEL AIR LOCK HOIST
1AH218, 1BH218	PERSONNEL AIR LOCK SHIELD REMOVAL HOISTS
1AH201, 1BH201	RECIRCULATION PUMP MOTOR HOISTS



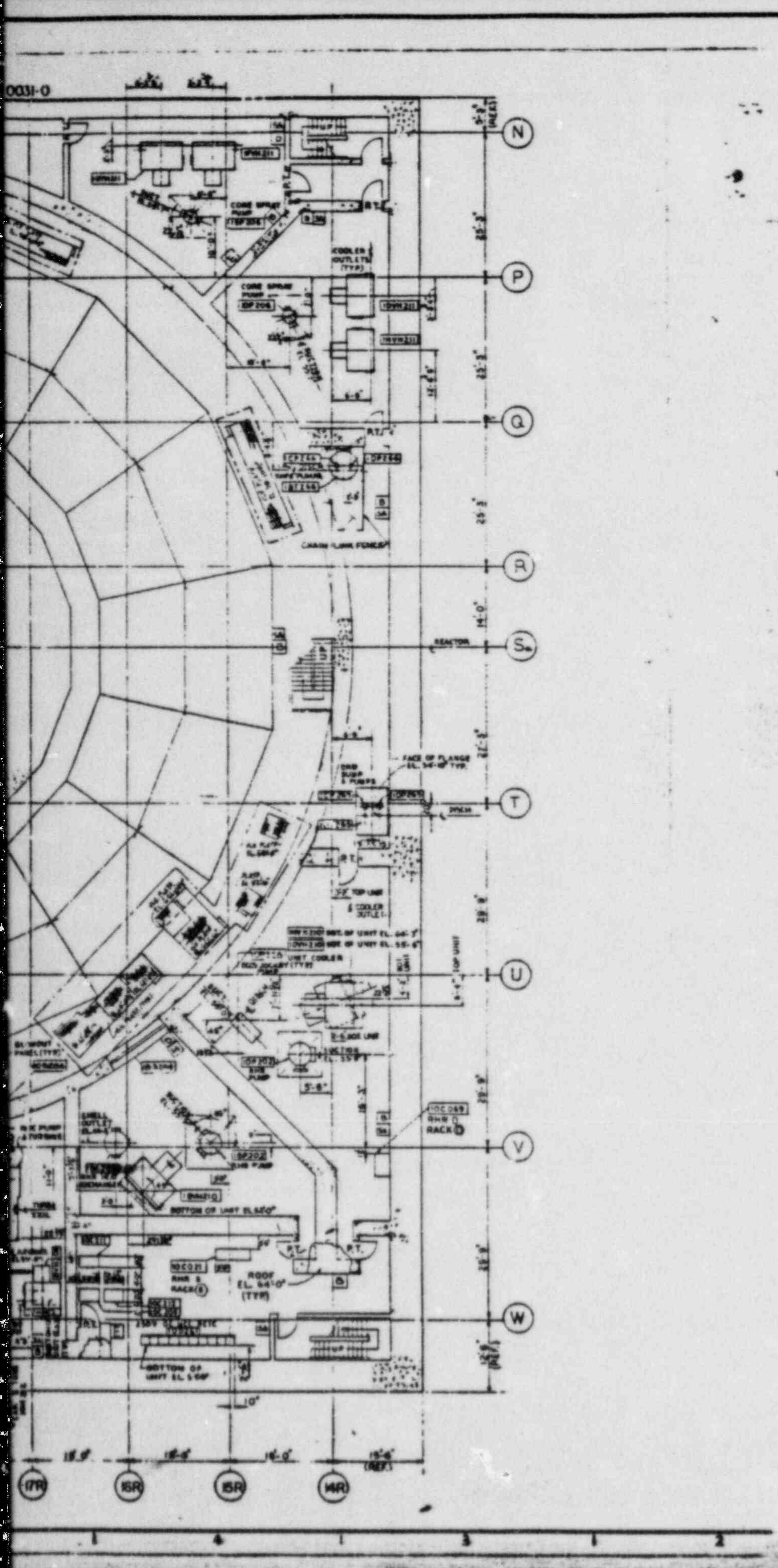
HOPE CREEK GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

SAFE LOAD PATH DRAWING (EL. 102) PERSONNEL AIR LOCK AND RECIRCULATION PUMP MOTOR HOISTS

FIGURE 9.1-33 — AMENDMENT 6, 10/84



MATCH LINE, FOR CONT. SEE DWG.
MATCH LINE, FOR CONT. SEE DWG. P-00510
TORUS
LOAD PATH FOR
HPCI PUMP AND
TURBINE PARTS
23R
22R
21R
20R
19R
18.9
18R



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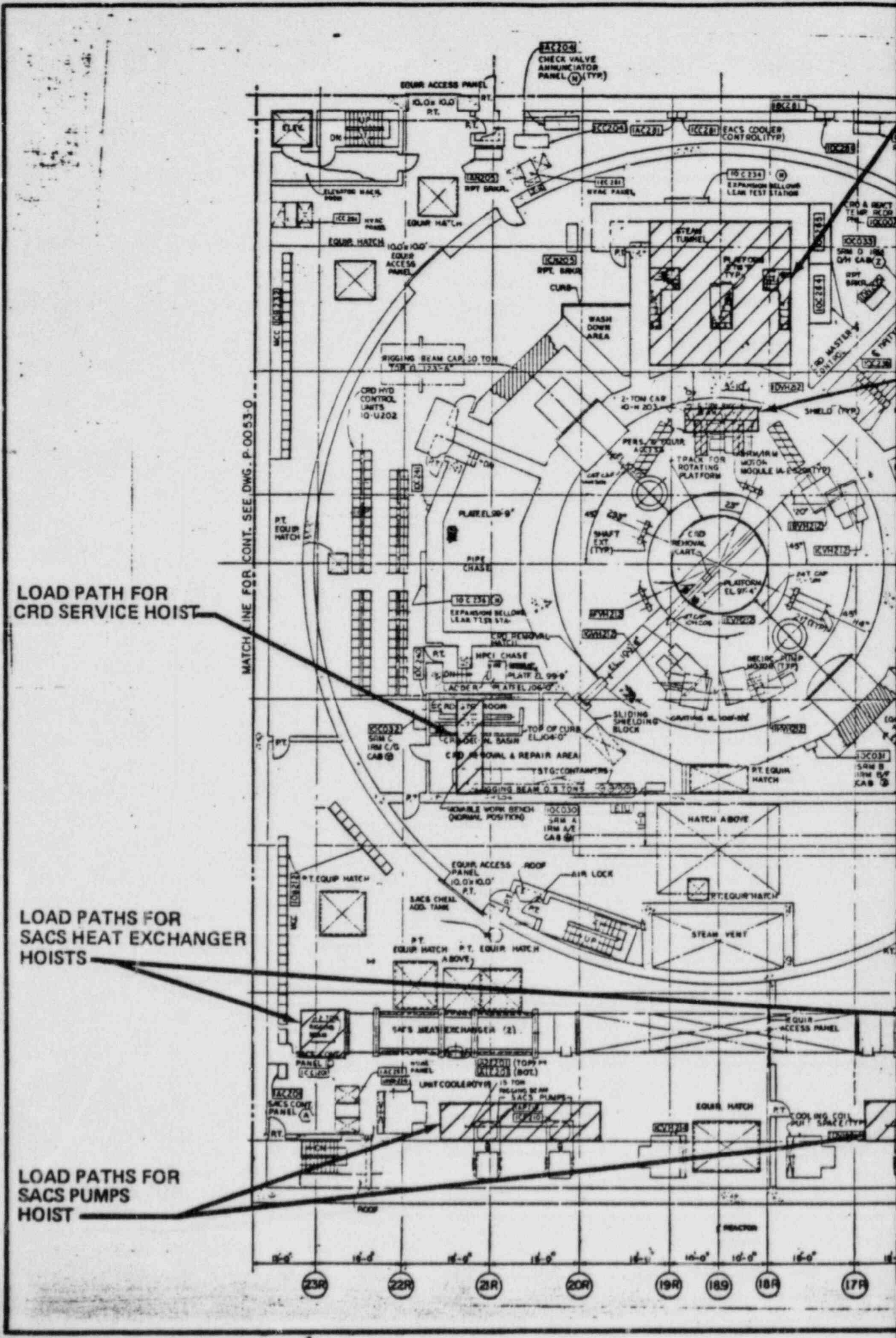
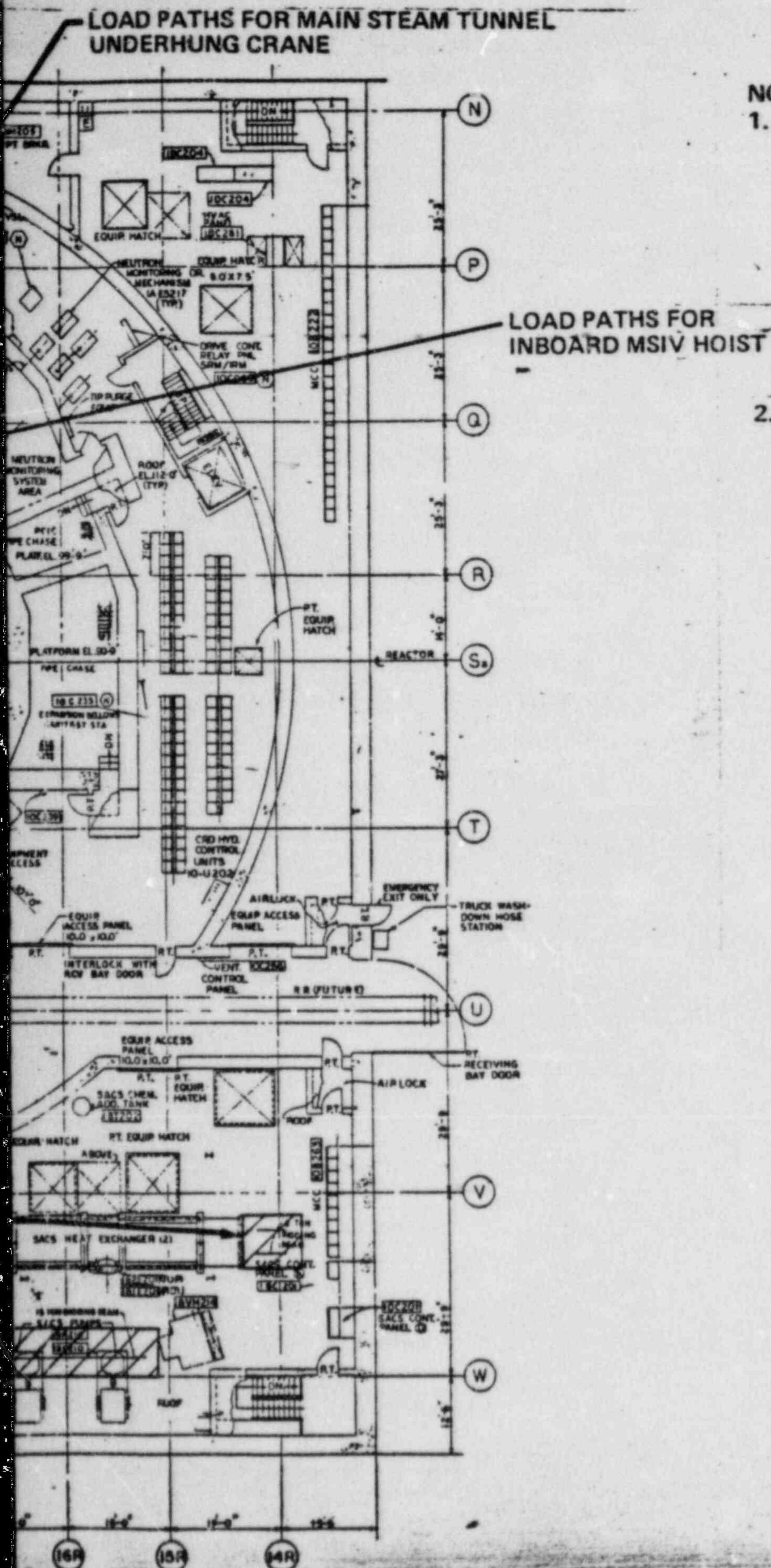
FINAL SAFETY ANALYSIS REPORT

SAFE LOAD PATH DRAWING (EL. 54)
HPCI PUMP AND TURBINE HOISTS
1AH211, 1BH211

FIGURE 9.1-36


LOAD PATH FOR
CRD SERVICE HOIST
LOAD PATHS FOR
SACS HEAT EXCHANGER
HOISTS
LOAD PATHS FOR
SACS PUMPS
HOIST
MATCH LINE FOR CONT. SEE DWG. P-0053-0
SACS HEAT EXCHANGER (2)
HATCH ABOVE
STEAM VENT
WASH DOWN AREA
STEAM TUNNEL
PIPE CHASE
CRD REMOVAL & REPAIR AREA
AIR LOCK
23R
22R
21R
20R
19R
18.9
18R
17R



NOTES:

1. THE LOAD PATHS ON THIS DRAWING ARE FOR:
   10H219 INBOARD MSIV HOIST
   10H214, 10H223 MAIN STEAM TUNNEL UNDERHUNG CRANE
   * CRD SERVICE HOIST
   * SACS PUMPS HOIST
   * SACS HEAT EXCHANGER HOISTS
2. HOIST DENOTED BY * WILL BE BORROWED FROM ANOTHER LOCATION WHEN NEEDED

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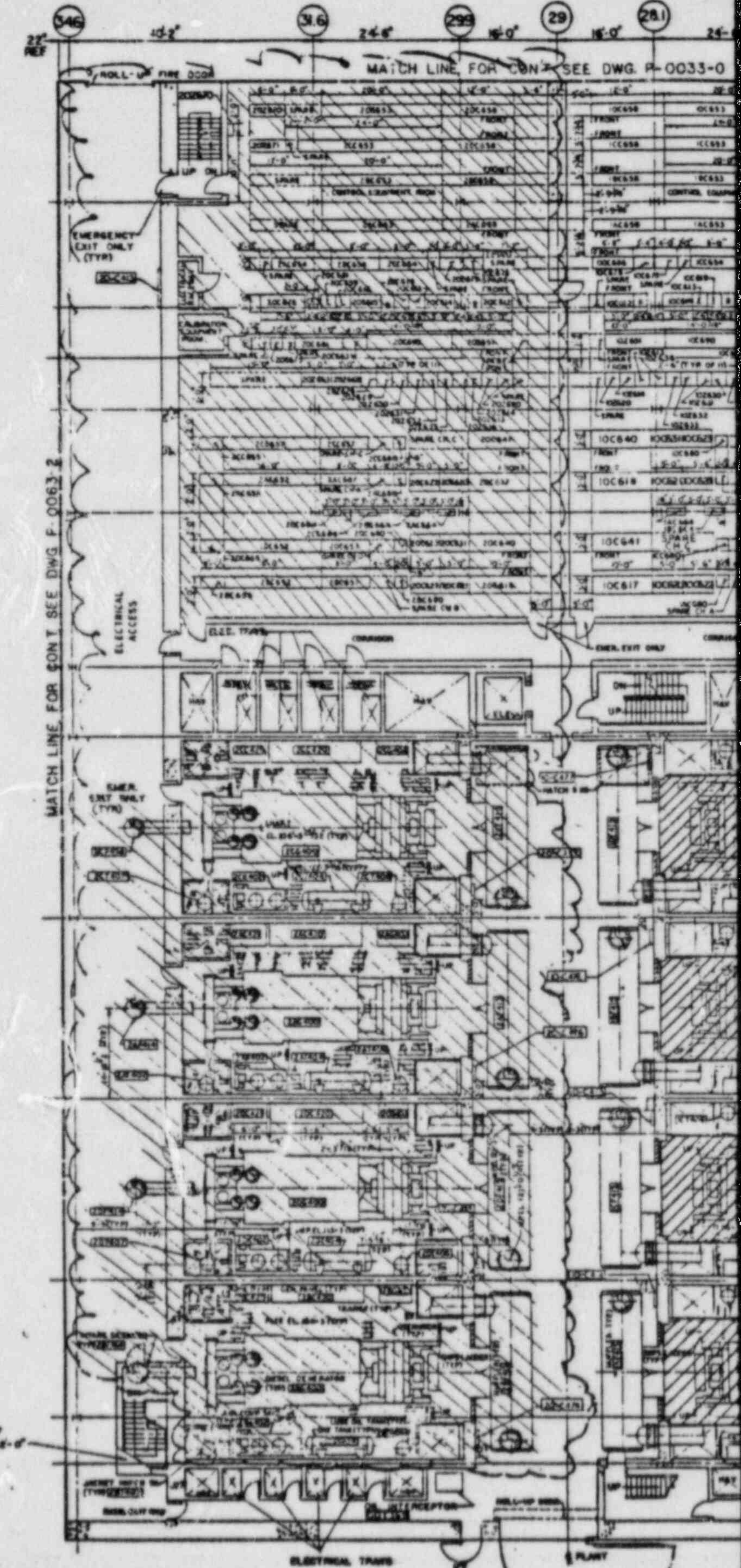
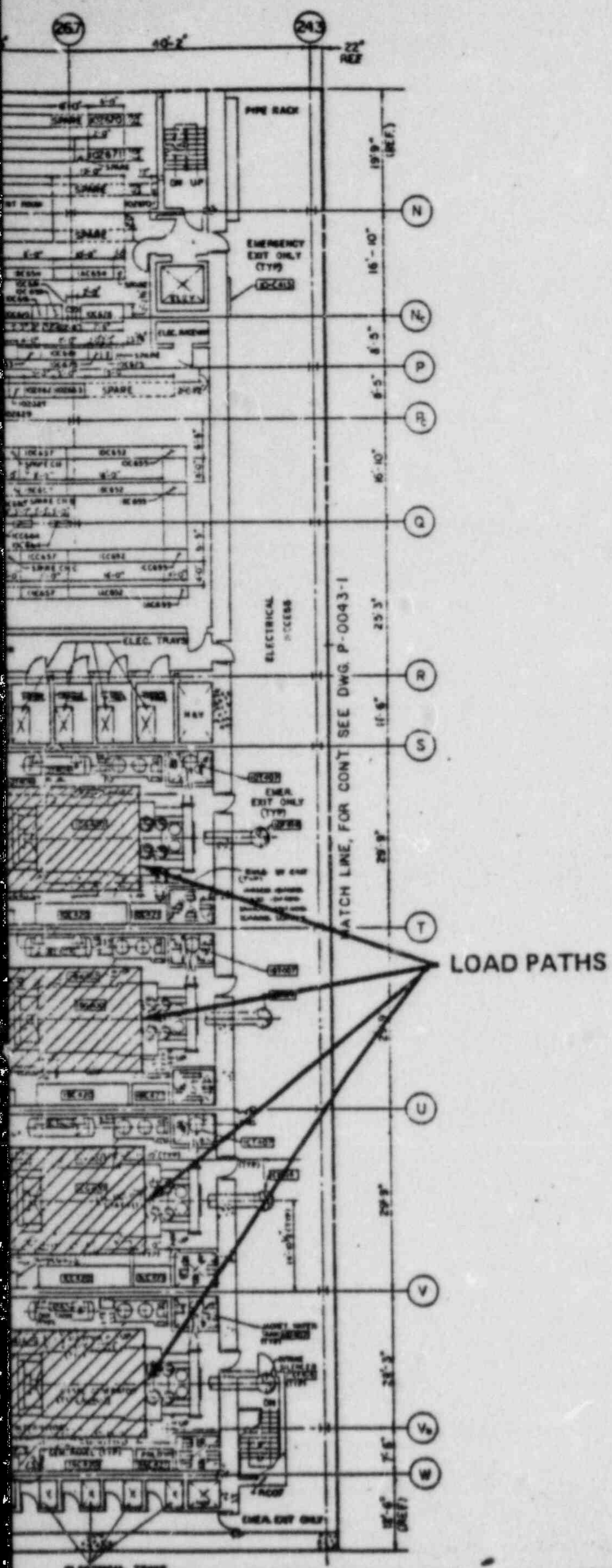
HOPE CREEK GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

SAFE LOAD PATH DRAWING (EL. 102)
TABLE 9.1-10, ITEMS 7, 8, 39, 40, 41

FIGURE 9.1-35 [illegible]



MATCH LINE FOR CON'T SEE DWG. P-0033-0
MATCH LINE FOR CON'T SEE DWG P-0063-2
ROLL-UP FIRE DOOR
EMERGENCY EXIT ONLY (TYP)
ELECTRICAL ACCESS
CONTROL EQUIPMENT ROOM
CORRIDOR
EMER. EXIT ONLY
ELECTRICAL TRAYS



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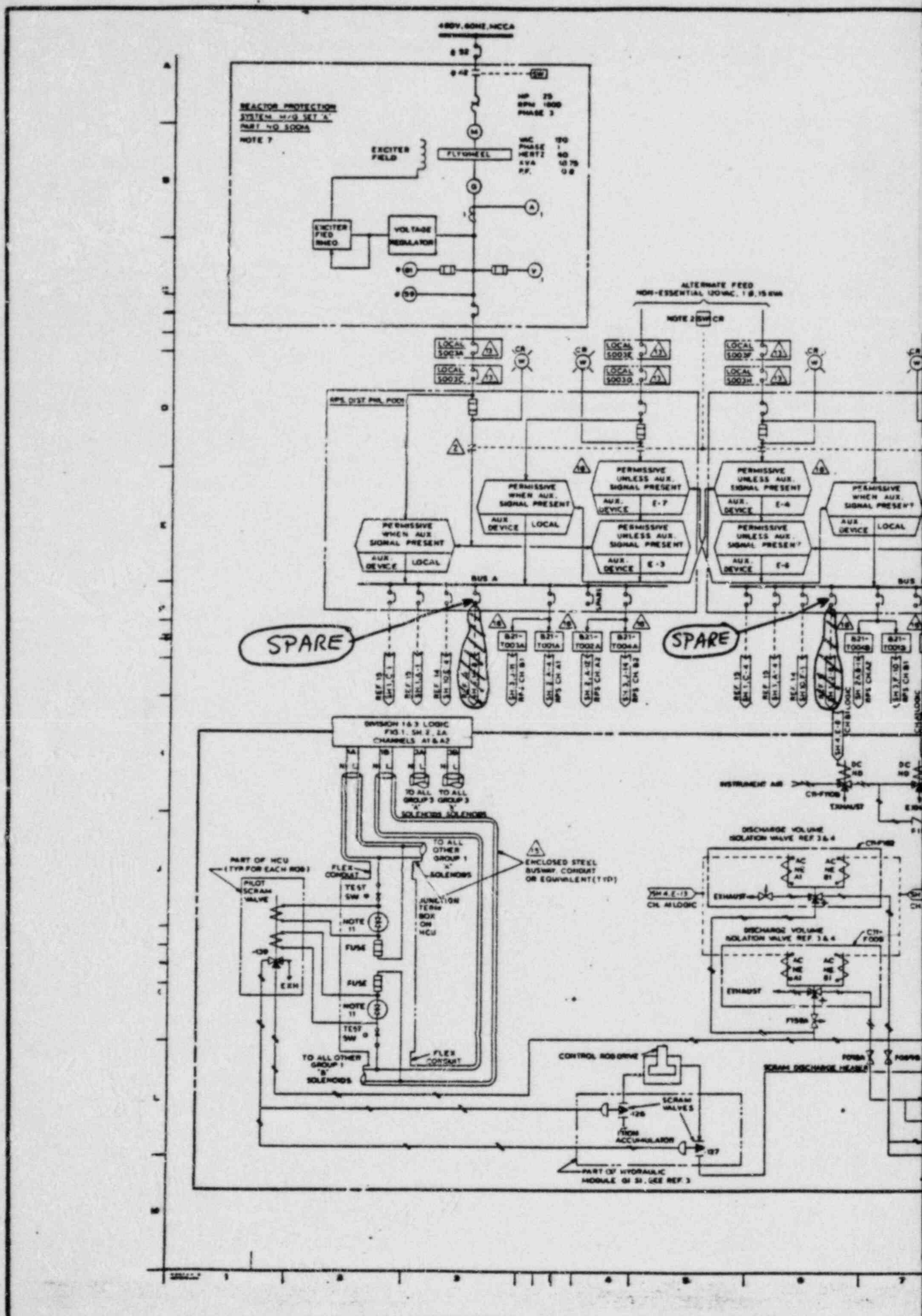
HOPE CREEK GENERATING STATION

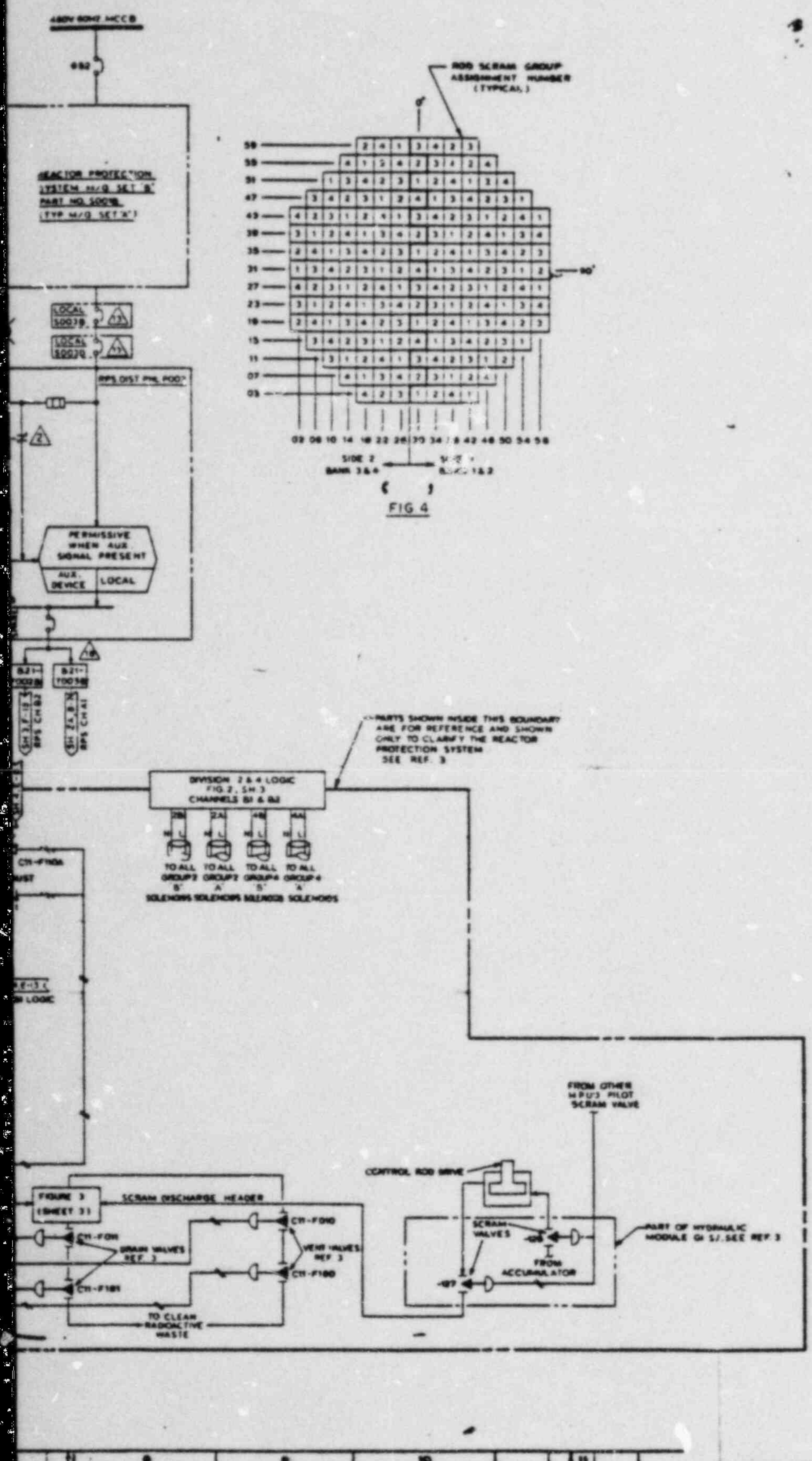
FINAL SAFETY ANALYSIS REPORT

SAFE LOAD PATH DRAWING (EL. 102)
DIESEL GENERATOR CRANES
1AH400, 1BH400, 1CH400, 1DH400

FIGURE 9.1-36



480V, 60HZ, MCCA
REACTOR PROTECTION SYSTEM M/G SET 'A' PART NO S00A
NOTE 7
EXCITER FIELD
FLYWHEEL
VOLTAGE REGULATOR
ALTERNATE FEED NON-ESSENTIAL 120VAC, 1 Ø, 15 KVA
RPS DIST PNL
PERMISSIVE WHEN AUX. SIGNAL PRESENT
PERMISSIVE UNLESS AUX. SIGNAL PRESENT
AUX. DEVICE
LOCAL
BUS A
SPARE
SPARE
DIVISION 1 & 3 LOGIC
TO ALL GROUP 3 SOLENOIDS
TO ALL OTHER GROUP 1 SOLENOIDS
ENCLOSED STEEL BUSWAY, CONDUIT OR EQUIVALENT (TYP)
PART OF HCU (TYP FOR EACH ROD)
PILOT SCRAM VALVE
FLEX CONDUIT
TEST SW
NOTE 11
FUSE
JUNCTION TERM BOX ON HCU
EXH
TO ALL OTHER GROUP 1 'B' SOLENOIDS
INSTRUMENT AIR
EXHAUST
DISCHARGE VOLUME ISOLATION VALVE REF 3 & 4
CH. A1 LOGIC
CONTROL ROD DRIVE
SCRAM VALVES
FROM ACCUMULATOR
PART OF HYDRAULIC MODULE
SCRAM DISCHARGE HEADER



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HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

REACTOR PROTECTION SYSTEM

SHEET 1 OF 5
FIGURE 7.2-1

AMENDMENT 5, 4/84

8409110175-22

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 →

~~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:~~

delete

~~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~~

DSER OPEN ITEM 166 (REV. 1)

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

DSER OPEN ITEM 166 (REV. 1)

<u>Insert</u>

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. INSERT A

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.

INSERT A

The location of portable monitors which will be positioned within the station to provide supplemental inplant monitoring of particulates and iodine levels will be provided by July 1, 1985. The positioning of supplemental continuous air monitors is part of the Radiation Protection Program and a July 1, 1985 date is consistent with finalizing other details of the program (i.e., instrument and equipment calibration). The location, quantity, and monitor type will be provided at that time.

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9/6/84
M P84 93/10 1

DSER OPEN ITEM 166 (REV.1)

QUESTION 421.10 (SECTION 7.1 & 7.2)

The staff believes that the physical separation provided in the design of the RPS cabinets may not satisfy the requirements of Regulatory Guid 1.75 or the plant separation criteria and is, therefore, unacceptable. As an example, it has been noted on similar plants that the cabinet lighting and power circuits (which are not treated as associated circuits) becomes associated with Class 1E circuits inside the RPS cabinets. Section 8.1.4.14 includes a brief discussion on the physical separation provided within panels, instrument racks and control boards for the instrumentation and control circuits of different divisions. Review the design of all Class 1E cabinets for separation between non-Class 1E and Class 1E circuits. Provide the staff with a listing of the cabinets which were reviewed and describe in detail how physical separation is maintained within the panels, racks and boards for those cases where a 6 inch air space cannot be maintained. Provide a summary of the analysis and testing performed to support this lesser separation. Include in the discussion the separation provided for associated circuits, internal wiring identification and the use of common terminations.

RESPONSE

The HCGS RPS cabinets (10C609, 10C611, 10C622 and 10C623) meet the requirements of IEEE Standard 384 as modified and endorsed by Regulatory Guide 1.75, as stated in Section 1.8.1.75. Cabinet lighting and receptacle power circuits are physically separated from RPS circuits by being routed in metallic conduit or by structural steel barriers.

Physical separation between non-Class 1E and Class 1E instrumentation and control circuits is provided in panels, instrument racks and control boards in accordance with IEEE Standard 384, as modified and endorsed by Regulatory Guide 1.75 as stated in Section 1.8.1.75. The following is a listing of Class 1E panels, instrument racks and control boards reviewed for the separation requirements of IEEE Standard 384:

Panels

1AC200	H_2/O_2 Analyzer A Panel
1BC200	H_2/O_2 Analyzer B Panel
1CC200	H_2/O_2 Analyzer Heat Trace Panel
1DC200	H_2/O_2 Analyzer Heat Trace Panel
1AC201	SACS Control Panel A
1BC201	SACS Control Panel B
1CC201	SACS Control Panel C
1DC201	SACS Control Panel D
10C202	RACS Heat Exchanger and Pumps Control Panel

1AC213	Instrument Gas Compressor A Control Panel
1BC213	Instrument Gas Compressor B Control Panel
1AC215	H_2 Recombiner A Power Distribution Panel
1BC215	H_2 Recombiner B Power Distribution Panel
1AC281	Reactor Building Unit Cooler Control Panel
1BC281	Reactor Building Unit Cooler Control Panel
1CC281	Reactor Building Unit Cooler Control Panel
1DC281	Reactor Building Unit Cooler Control Panel
1AC285	Reactor Building FRVS Control Panel
1BC285	Reactor Building FRVS Control Panel
1CC285	Reactor Building FRVS Control Panel
1DC285	Reactor Building FRVS Control Panel
1OC286	Reactor Building Equipment Lock Ventilation
1OC399	Remote Shutdown Panel
1OC401	Diesel Generator Area Battery Room Panel
1OC402	Diesel Generator Area Battery Room Panel
1AC420	Diesel Generator A Exciter Panel
1BC420	Diesel Generator B Exciter Panel
1CC420	Diesel Generator C Exciter Panel
1DC420	Diesel Generator D Exciter Panel
1AC421	Diesel Generator A Local Engine Control Panel
1BC421	Diesel Generator B Local Engine Control Panel
1CC421	Diesel Generator C Local Engine Control Panel
1DC421	Diesel Generator D Local Engine Control Panel
1AC422	Diesel Generator A Remote Control Generator Panel
1BC422	Diesel Generator B Remote Control Generator Panel
1CC422	Diesel Generator C Remote Control Generator Panel
1DC422	Diesel Generator D Remote Control Generator Panel
1AC423	Diesel Generator A Remote Engine Control Panel
1BC423	Diesel Generator B Remote Engine Control Panel
1CC423	Diesel Generator C Remote Engine Control Panel
1DC423	Diesel Generator D Remote Engine Control Panel
1AC428	Diesel Generator A Load Sequencer Panel
1BC428	Diesel Generator B Load Sequencer Panel
1CC428	Diesel Generator C Load Sequencer Panel
1DC428	Diesel Generator D Load Sequencer Panel
1AC482	Electric Heater Control Panel 1AVH403
1BC482	Electric Heater Control Panel 1BVH403
1AC483	Diesel Area HVAC Control Panel
1BC483	Diesel Area HVAC Control Panel
1CC483	Diesel Area HVAC Control Panel
1DC483	Diesel Area HVAC Control Panel
1AC485	Control Area HVAC Control Panel
1BC485	Control Area HVAC Control Panel
1AC486	Diesel Area Panel Room Supply System
1BC486	Diesel Area Panel Room Supply System
1AC487	Water Chiller Panel
1BC487	Water Chiller Panel
1AC488	Chiller AK403 Power Panel
1BC488	Chiller BK403 Power Panel
1AC489	Electric Heater Control Panel 1AVH407
1BC489	Electric Heater Control Panel 1BVH407

1AC490	Water Chiller A Control Panel
1BC490	Water Chiller B Control Panel
1AC491	Water Chiller A Power Panel
1BC491	Water Chiller B Power Panel
1AC492	Electric Heater Control Panel
1BC492	Electric Heater Control Panel
1AC493	Control Panel - Auxiliary Building Diesel
1AC494	Control Panel - Auxiliary Building Diesel
1AC495	Control Panel - Auxiliary Building Diesel
1BC495	Control Panel - Auxiliary Building Diesel
1CC495	Control Panel - Auxiliary Building Diesel
1DC495	Control Panel - Auxiliary Building Diesel
1AC515	Traveling Screen Control Panel
1BC515	Traveling Screen Control Panel
1CC515	Traveling Screen Control Panel
1DC515	Traveling Screen Control Panel
1AC516	Service Water Pump Panel
1BC516	Service Water Pump Panel
1CC516	Service Water Pump Panel
1DC516	Service Water Pump Panel
1AC581	Intake Structure HVAC Control Panel
1BC581	Intake Structure HVAC Control Panel
1CC581	Intake Structure HVAC Control Panel
1DC581	Intake Structure HVAC Control Panel
10C601	RRCS Division 1 Panel
10C602	RRCS Division 2 Panel
10C604	Class 1E Radiation Monitoring Instrumentation Cabinet
10C617	Division 1 RHR and Core Spray Relay Vertical Board
10C618	Division 2 RHR and Core Spray Relay Vertical Board
10C620	HPCI Relay Vertical Board
10C621	RCIC Relay Vertical Board
10C622	Inboard Isolation Valve Relay Vertical Board
10C623	Outboard Isolation Valve Relay Vertical Board
10C628	ADS Division 2 Relay Vertical Board
10C631	ADS Division 4 Relay Vertical Board
1AC633	Post LOCA H_2 Recombiner A Control Cabinet
1BC633	Post LOCA H_2 Recombiner B Control Cabinet
10C640	Division 4 RHR and Core Spray Relay Vertical Board
10C641	Division 3 RHR and Core Spray Relay Vertical Board
10C650	Main Control Room Vertical Board
10C651	Unit Operators Console
1AC652	1E Solid State Logic Cabinet Channel A
1BC652	1E Solid State Logic Cabinet Channel B
1CC652	1E Solid State Logic Cabinet Channel C
1DC652	1E Solid State Logic Cabinet Channel D
1AC655	1E Analog Logic Cabinet Channel A
1BC655	1E Analog Logic Cabinet Channel B
1CC655	1E Analog Logic Cabinet Channel C
1DC655	1E Analog Logic Cabinet Channel D
1AC657	1E Digital Termination Cabinet Channel A
1BC657	1E Digital Termination Cabinet Channel B
1CC657	1E Digital Termination Cabinet Channel C

1DC657	1E Digital Termination Cabinet Channel D
1AC680	1E Electrical Auxiliary Cabinet Channel A
1BC680	1E Electrical Auxiliary Cabinet Channel B
1CC680	1E Electrical Auxiliary Cabinet Channel C
1DC680	1E Electrical Auxiliary Cabinet Channel D

Instrument Racks

10C002	Reactor Water Clean-up Rack
10C004	Reactor Vessel Level and Pressure A Rack
10C005	Reactor Vessel Level and Pressure C Rack
10C009	Jet Pump Rack A
10C014	HPCI A/HPCI Leak Detection A Rack
10C015	Main Steam C/D and Recirc A Flow Rack
10C018	RHR A and ADS Rack
10C021	RHR B and ADS Rack
10C025	Main Steam C/D and Recirc A Flow Rack
10C026	Reactor Vessel Level and Pressure D Rack
10C027	Reactor Vessel Level and Pressure B Rack
10C037	RCIC D/RCIC Leak Detection D Rack
10C041	Main Steam A/B and Recirc B Flow Rack
10C042	Main Steam A/B and Recirc B Flow Rack
10C069	RHR D and ADS Rack
10C208A	RCIC/Reactor Cooling
10C211	RCIC Pump
10C212	RCIC Pump

Instrument racks are separated into channels. No two redundant piped or tubed safety-related instruments are located on the same rack.

Where a 6-inch air space cannot be maintained between instrumentation and control circuits of different channels (both Class 1E to Class 1E and Class 1E to non-Class 1E), barriers are provided in accordance with IEEE Standard 384. These barriers are metallic conduit, structural steel barriers, or non-metallic wrap (Havey Industries Siltemp Sleeving Type S or Siltemp Woven Tape Type WT65). The metallic conduit and structural steel barriers are noncombustible materials. The nonmetallic wrap (Siltemp) was successfully tested for use as an isolation barrier (reference Wyle Laboratories Test Report Number 56669).

~~For certain types of isolation devices, barriers of the type noted above are not feasible. For these cases, requirements of Section 7.2.2.1 of IEEE Standard 384 are met, as follows:~~

> ~~"The separation of the wiring at the input and output terminals of the isolation device may be less than 6 inches (0.15 m) as required in 6.6.2 provided that it is not less than the distance between input and output terminals.~~

Add Insert A

- INSERT A -

At HCGS, three isolation devices are used which do not satisfy the 6 inch air space requirement and, by design, barriers of the type identified above are not feasible. The 6 inch air space requirement is maintained for wiring associated with these devices except at the device itself where the separation is maintained not less than the physical distance between the input and output terminals of the isolation device. These devices are:

a) TEC analog isolator, model 156 - provides class 1E to non-class 1E isolation for low level analog inputs to the plant computer,

b) Struthers Dunn type 219 relay - provides class 1E to non-class 1E isolation for inputs to the plant annunciator (125 vdc contact interrogation voltage is used by the plant annunciator),

c) Allen Bradley model 700-200A12P relay - provides class 1E to non-class 1E isolation for inputs to the plant annunciator.

These devices are fully qualified for their application as described in part (d) of the response to question 421.12

~~Minimum separation requirements do not apply for wiring and components within the isolation device; however, separation shall be provided wherever practicable."~~

~~Testing, in accordance with IEEE Standard 472 (Surge Withstand Capability) will be performed to ensure that the Class 1E inputs to the isolation devices are not affected by short-circuit failures, ground faults or voltage surges on the output side of the isolation devices.~~

Single Failure ~~The only~~ analysis were ~~that will be~~ performed to support air spaces less than 6 inches, ~~since the requirements of IEEE Standard 384 are satisfied, is~~ for the Neutron Monitoring System Panel (10C608) and the Process Radiation Monitoring System Panels (10C635 and 10C636). This report was submitted under separate cover (R.L. Mittl to A. Schwencer dated September 7, 1984) No associated circuits have been identified in the non-NSSS panels, instrument racks, or control boards. Internal wiring identification is done using color coded insulation or insulation marked with color coded tape. For panel sections of one channel only, internal wiring identification may not be done. Where common terminations are used, the requirements of IEEE Standard 384 are satisfied as stated above.

Electrical equipment and wiring for the reactor protection system (RPS), the nuclear steam supply shutoff systems (NSSSS) and the engineered safeguards subsystems (ESS) are segregated into separate divisions designated I and II, etc., such that no single credible event is capable of disabling sufficient equipment to prevent reactor shutdown, removal of decay heat from the core, or closure of the NSSSS valves in the event of a design basis accident.

No single control panel section (or local panel section or instrument rack) includes wiring essential to the protective function of two systems that are backups for each other (Division I and Division II) except as allowed below:

a. If two panels containing circuits of different separation divisions are less than 3 feet apart, there shall be a steel barrier between the two panels. Panel ends closed by steel end plates are considered to be acceptable barriers provided that terminal boards and wireways are spaced a minimum of one inch from the end plate.

b. Floor-to-panel fire proof barriers must be provided between adjacent panels having closed ends.

c. Penetration of separation barriers within a subdivided panel is permitted, provided that such penetrations are sealed or otherwise treated so that an electrical fire could not reasonably propagate from one section to the other and destroy the protective function.

d. Where, for operational reasons, locating manual control switches on separate panels is considered to be prohibitively (or unduly) restrictive to normal functioning of equipment, then the switches may be located on the same panel provided no single event in the panel can defeat the automatic operation of the equipment.

With the exception of panels 10C608, 10C635 and 10C636, internal wiring of the NSSS panels and racks has color-coded insulation. Associated circuits are treated within a panel or rack in the same manner as the essential circuits. Where common terminations are used, the requirements of IEEE Standard 384 are satisfied.

Electrical protection assemblies have been added between the power range NMS panel (10C608) and its two 120V ac UPS power feeders as described in revised Section 7.6.1.4.2.

CHAPTER 7

FIGURES (Cont)

coaxial cable. The amplifier is a linear current amplifier whose voltage output is proportional to the current input and therefore proportional to the magnitude of the neutron flux. Low level output signals are provided that are suitable as an input to the computer, recorders, etc. The output of each LPRM amplifier is isolated to prevent interference of the signal by inadvertent grounding or application of stray voltage at the signal terminal point.

Power for the LPRM is supplied ~~by~~ from two non-Class 1E uninterruptible power sources (see Figure 8.3-11, Sht. 3). Approximately half of the LPRMs are supplied from each bus. Each LPRM amplifier has a separate power supply in the main control room, which furnishes the detector polarizing potential. The LPRM amplifier cards are mounted into pages in the NMS cabinet, and each page is supplied operating voltages from a separate low voltage power supply.

- INSERT A -

The trip circuits for the LPRM provide signals to actuate lights and annunciators. Table 7.6-3 lists the LPRM trips.

Each LPRM may be individually bypassed via a switch on the LPRM amplifier card. Placing an LPRM in "bypass" sends a signal to the assigned APRM, electronically causing it to adjust its averaging amplifier's gain to allow for one less LPRM input. In this way, each APRM can continue to produce an accurate signal representing average core power even if some of the assigned LPRMs fail during operation. If the number of functional assigned LPRMs drops to 50% of the normal number, the APRM automatically goes inoperative and a half scram (one trip logic channel deenergized), rod block, and appropriate annunciation are generated. Administrative controls ensure that a minimum number of LPRMs at each level (A, B, C, and D) in the core are maintained or the APRM is declared inoperative and manually placed in the tripped state.

In addition to the signals supplied to the APRMs, the LPRMs also send flux signals to the rod block monitor (RBM). When a central control rod is selected for movement, the output signals from the amplifiers associated with the nearest 16 LPRM detectors are displayed on the main control room vertical board meters and sent to the RBM. The four LPRM detector signals from each of the four detector assemblies are displayed on 16 separate meters. The operator can readily obtain readings from all the LPRM detectors by selecting the control rods in order. These signals from the

INSERT A

Electrical protection assemblies (EPAs) identical to those used in the reactor protection system (RPS) (described in Section 8.3.1.5.4) are installed between the power range NMS and the two 120V AC feeders from the UPS power sources (see Figure 7.6-11). The EPAs ensure that the power range NMS never operates under degraded bus voltage or frequency conditions (undervoltage, overvoltage, underfrequency). The power range NMS panel (10C608) was analyzed with this power supply configuration to ensure that no single failure of the power range NMS could inhibit the proper operation of the reactor protection system or any other safety system required for the safe operation of the plant. The interfaces between the power range NMS and the RPS have adequate provisions for separation. The RPS cabling external to the NMS panel conforms to the separation guidelines of Regulatory Guide 1.75, which the RPS must satisfy. Within the panel, where the cable and wiring runs to the different RPS divisions do not conform to the Regulatory Guide 1.75 separation criteria, fire-resistant "Sil-Temp" tape is wrapped around the cables and wires. This eliminates the possibility of fault propagation between the RPS divisions. In accordance with paragraph 5.6.2 of IEEE Standard 384, this tape has been demonstrated to be acceptable.

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

7.6.1.4.3 Average Power Range Monitor Subsystem

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

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

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

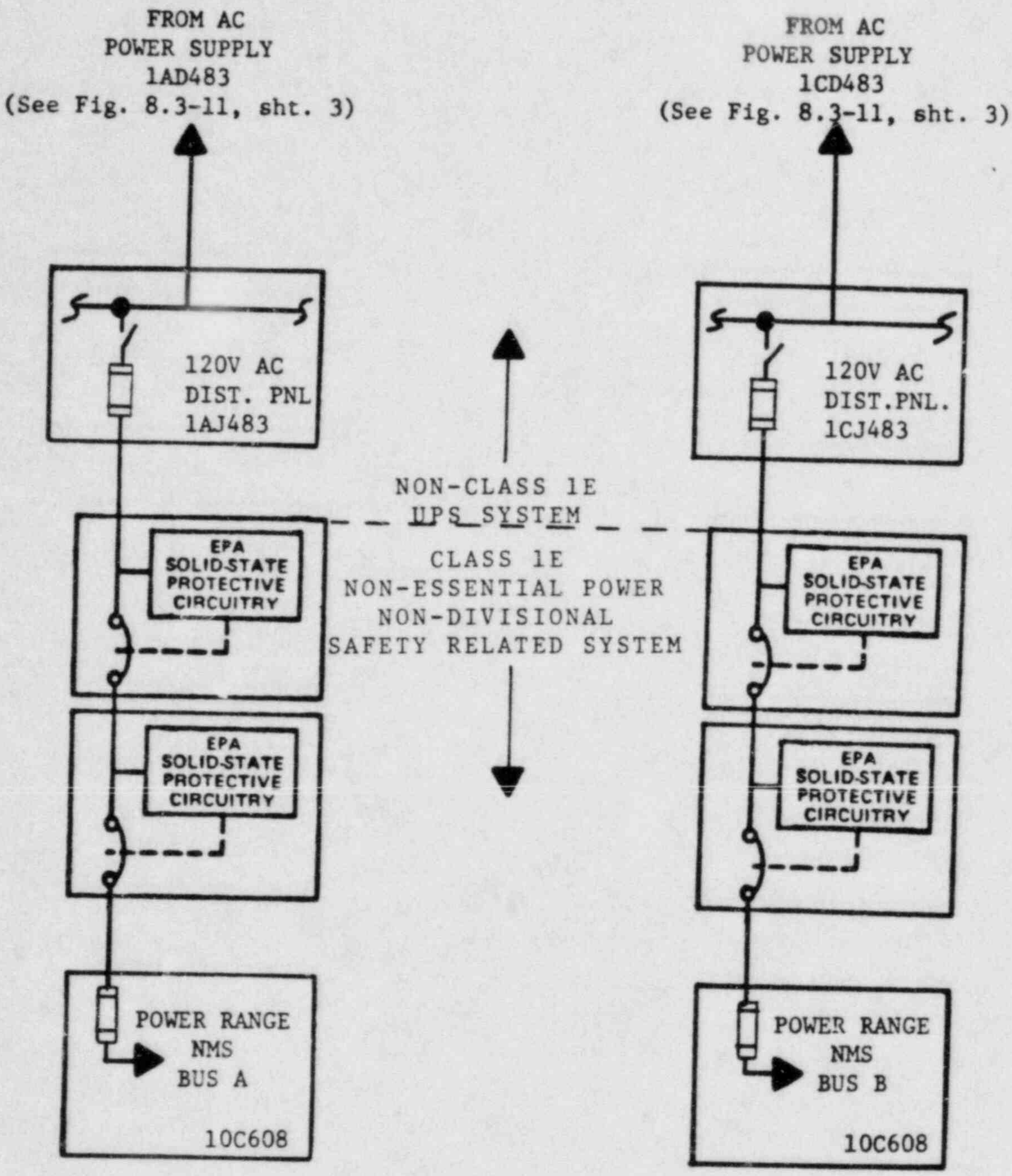
The APRM scram units are set for a reactor scram at 15% core power in "refuel" and "startup" modes. When the mode switch is in "run," the APRM trip reference signal is provided by a signal that varies with recirculation flow. This provides a power following reactor scram setpoint. As power increases,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

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

the same

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

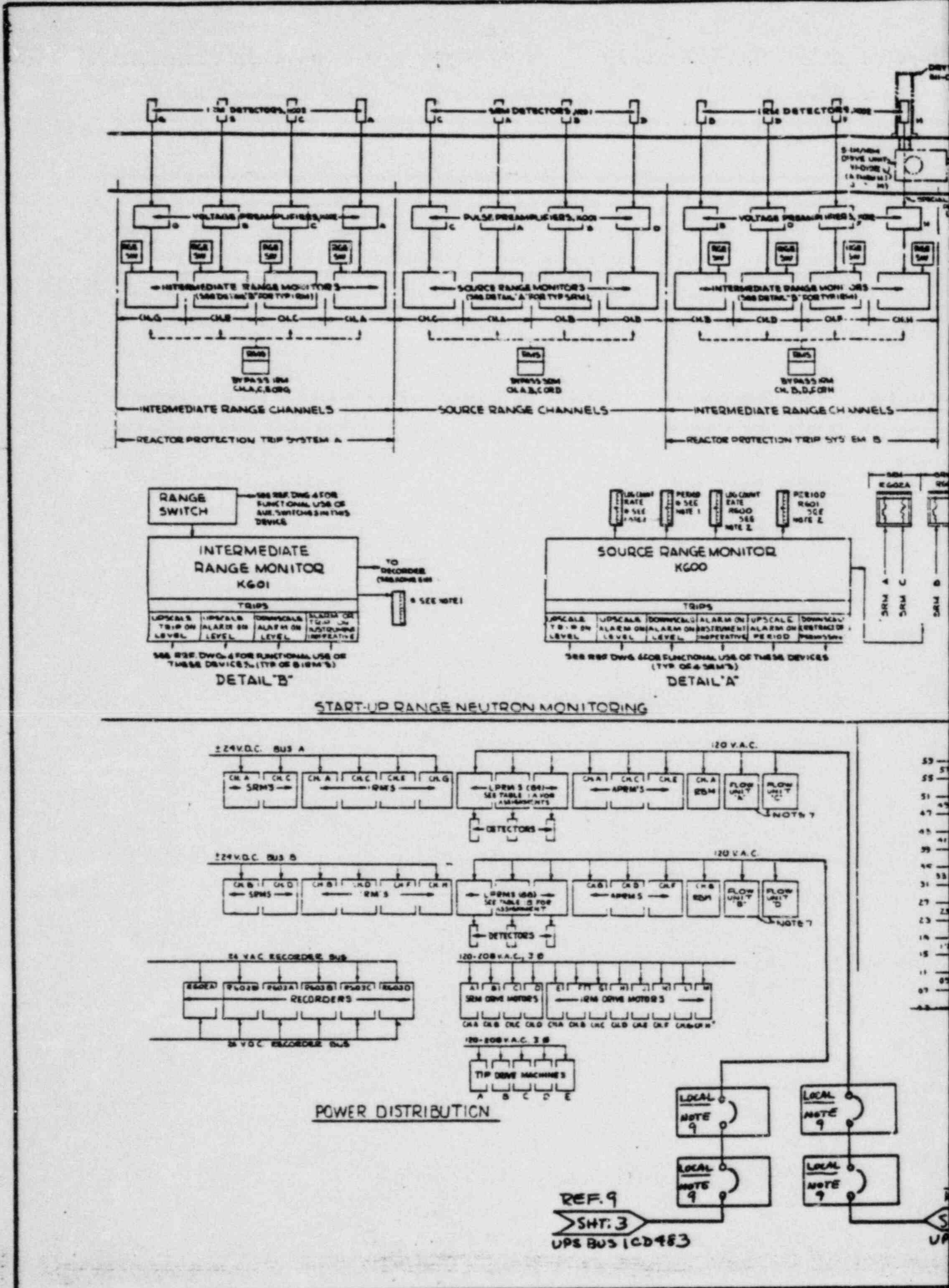
that supply the LPRMs (See Section 7.6.1.4.2).



HOPE CREEK
GENERATING STATION
FINAL SAFETY ANALYSIS REPORT

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

FIGURE 7.6-11 Amendment

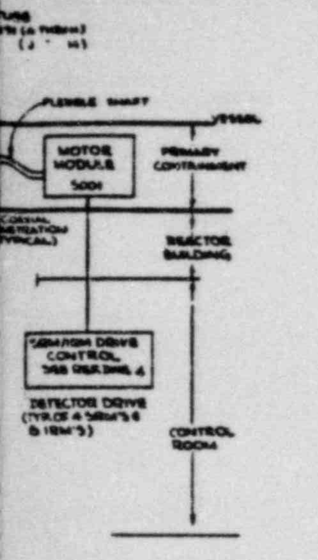


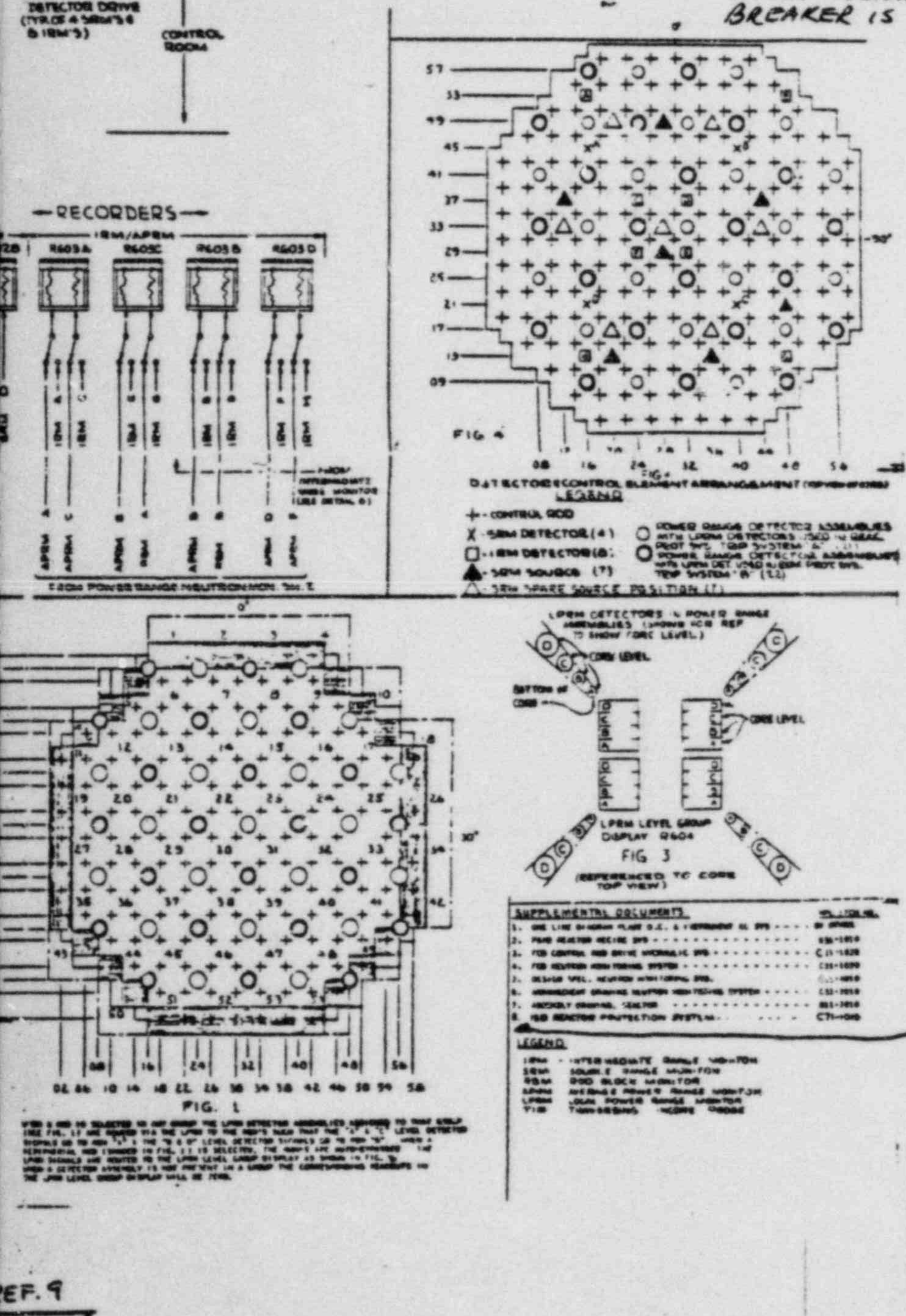
INTERMEDIATE RANGE MONITORS
SOURCE RANGE MONITORS
INTERMEDIATE RANGE CHANNELS
SOURCE RANGE CHANNELS
REACTOR PROTECTION TRIP SYSTEM A
RANGE SWITCH
INTERMEDIATE RANGE MONITOR K601
TRIPS
UPSCALE TRIP ON LEVEL
SEE NOTE 1
DETAIL "B"
SOURCE RANGE MONITOR K600
DETAIL "A"
START-UP RANGE NEUTRON MONITORING
±24 V.D.C. BUS A
120 V.A.C.
SRM'S
IRM'S
APRM'S
RBM
DETECTORS
NOTE 7
±24 V.D.C. BUS B
24 V.A.C. RECORDER BUS
RECORDERS
24 V.D.C. RECORDER BUS
120-208 V.A.C. 3 Ø
SRM DRIVE MOTORS
IRM DRIVE MOTORS
TIP DRIVE MACHINES
A B C D E
POWER DISTRIBUTION
LOCAL
NOTE 9
REF. 9
SHT. 3
UPS BUS 1CD483

NOTES:

8. EXCEPT FOR PART NO. 10, THE EXACT ASSIGNMENT OF TIP GUIDE TUBES FROM SPECIFIC INDEXING MECHANISMS TO SPECIFIC POWER RANGE DETECTOR ASSEMBLIES IN RESPECTIVE GROUPS SHOWN IS DETERMINED BY OTHERS AND WILL BE DELINEATED LATER.

9. CLASS IE ELECTRICAL PROTECTION ASSEMBLY TRIPS CIRCUIT BREAKER ON OVER-VOLTAGE, UNDER-VOLTAGE, OR UNDER-FREQUENCY. CIRCUIT BREAKER IS MANUALLY RESET.





9. SINGLE LINE ELECTRICAL DRAWING E-0012-1

Also Available On Aperture Card

TI APERTURE CARD

8409110175-23

729E615AC

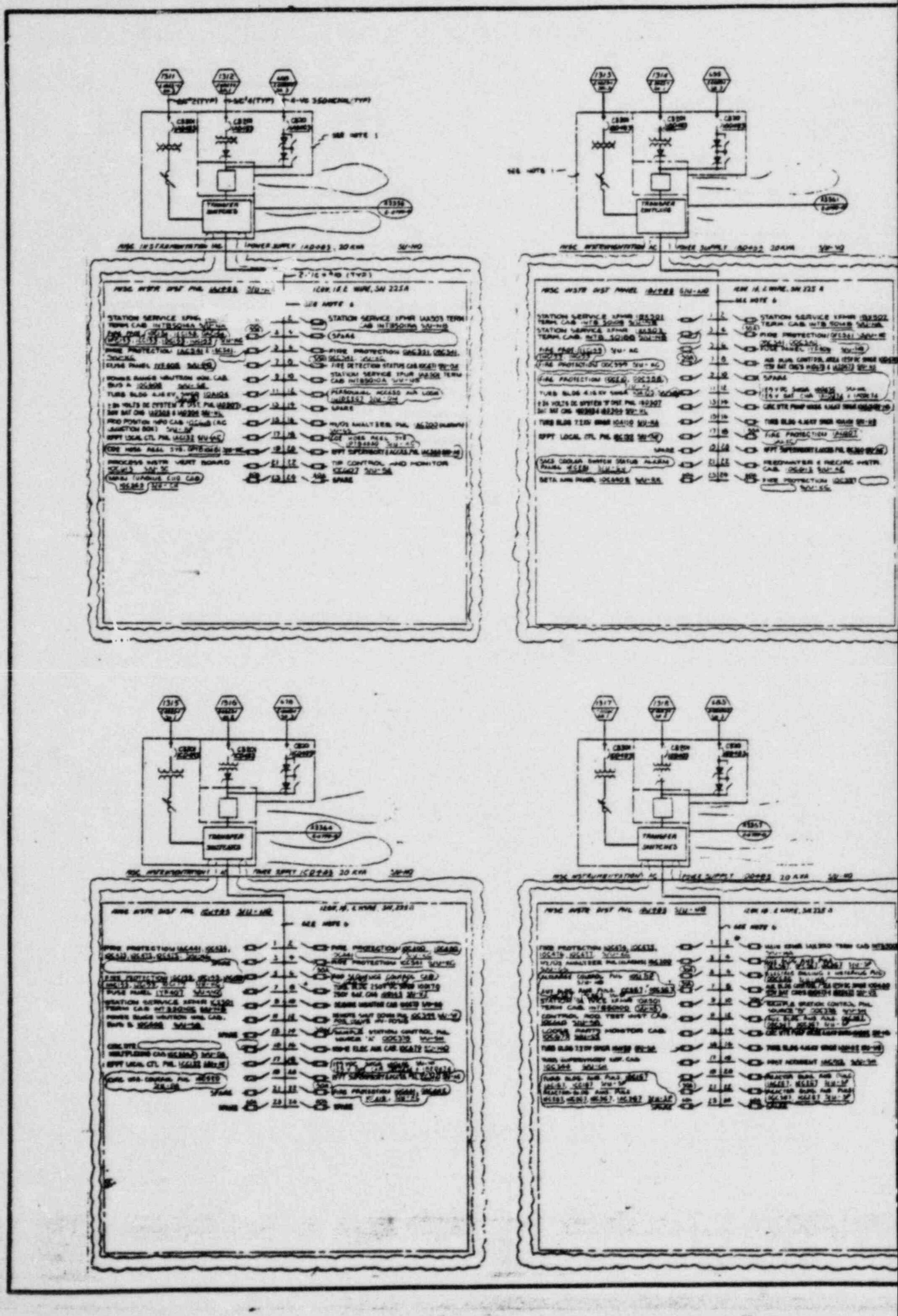
REF. 9
SHT. 3
BUS 1AD483

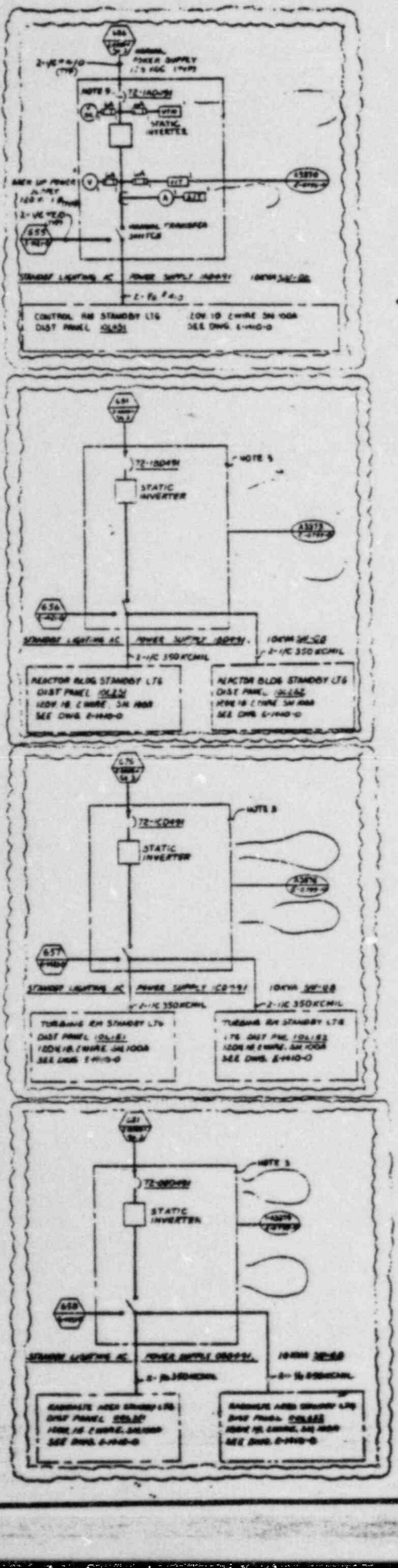
HOPE CREEK GENERATING STATION

FINAL SAFETY ANALYSIS REPORT

NMS IED

FIGURE 7.6-2
SHEET 1 OF 2


TRANSFER SWITCHES
STATION SERVICE XFMR
FIRE PROTECTION
SPARE
TIP CONTROL AND MONITOR
SEE NOTE 1
SEE NOTE 6



DEVICE TAG	DESCRIPTION	MFR/TYPE	REMARKS
AIT	CURRENT TRANSDUCER	[illegible]	
VT	VOLTAGE TRANSDUCER	[illegible]	
VT/F	DC VOLTAGE TRANSDUCER	[illegible]	
V	AC VOLTMETER	GE-AB40	
A	AC AMMETER	GE-AB40	
VDC	DC VOLTMETER	GE-DB40	
ADC	DC AMMETER	GE-DB40	

NOTES:

1. FOR MORE WIRING DETAILS SEE CHANNEL A POWER SUPPLY 1AD481, SHEET 1
2. WHENEVER THE RECTIFIER OUTPUT VOLTAGE OF THE NORMAL POWER SUPPLY GOES BELOW THE ALTERNATE POWER SUPPLY VOLTAGE LEVEL, THE LATTER POWER SUPPLY FEEDS THE INVERTER AND VICE VERSA. SHOULD BOTH NORMAL POWER AND ALTERNATE POWER OR ANY MAJOR COMPONENTS OR INVERTER FAIL, LOAD IS AUTOMATICALLY SWITCHED OVER TO A BACK-UP POWER SUPPLY WITHIN [illegible] AND WITHOUT LOSS OF POWER CONTINUITY TO THE LOAD. WHEN NORMAL POWER IS RESTORED, LOAD IS RETRANSFERRED TO IT AUTOMATICALLY WITH A TIME DELAY AFTER THE LOGIC CHECKS FOR SYNCHRONISM OF PHASES & VOLTAGE BALANCE BETWEEN THE TWO SOURCES.
3. FOR DETAILS SEE STDBY LTG AC POWER SUPPLY 1AD491.
4. FOR STANDBY LIGHTING, NORMAL POWER SUPPLY FROM THE STATION DC SYSTEM IS FED TO LOAD THROUGH THE INVERTER AND MANUAL TRANSFER SWITCH WHICH ALSO CONNECTS THE STATION AC SYSTEM AS BACK-UP POWER DURING INVERTER MALFUNCTION AND MAINTENANCE.
5. UNLESS OTHERWISE SPECIFIED ALL BRANCH CIRCUIT FUSES ARE 20 AMPS (FOR CABLE INFORMATION SEE REF. DWG. 1)
6. PANELBOARD NEUTRAL BUS SHALL BE GROUNDED
7. FOR ELECTRICAL NUMBERING SYSTEM SEE DWG. E-1410-0
8.
9. BREAKER WITH SHUNT TRIP ELEMENT ONLY IS TRIPPED AUTOMATICALLY BY A SIGNAL FROM THE CURRENT LIMIT SENSOR IN THE INVERTER AFTER A PRESET TIME DELAY.
10. DELETED
11. INVERTERS 1AD483, 1BD483, 1CD483 & 1DD483 ARE PURCHASED AS "Q" ITEMS BUT USED FOR "NON-Q" APPLICATION.

Also Available On Aperture Card

ADDITIONAL SYMBOLS:

— REF NUMBER
— CONTINUATION DWG. & SHEET NUMBER

— COMPUTER ID NUMBER
— SCHEME DWG. NUMBER
— CIRCUIT BKR [illegible] TYPE [illegible] (TYPICAL)

SU-HQ — START UP SYSTEM [illegible]

SN — SOLID NEUTRAL

EQUIPMENT LOCATION NUMBER:

1AD483 MISC INSTRUMENT AC POWER SUPPLY

TI APERTURE CARD

REFERENCE DRAWINGS:

1. [illegible] CABLE BLOCK DIAGRAM
2. VENDOR PRINT [illegible]
3. VENDOR PRINT [illegible]

ALL EQUIPMENT ON THIS SHEET IS NON-Q LISTED

8409110175-24

E-0012-1

REV 5

HOPE CREEK GENERATING STATION

FINAL SAFETY ANALYSIS REPORT

SINGLE LINE METER & RELAY DIAGRAM 120 V AC INSTRUMENTATION & MISC SYSTEMS

FIGURE 8.3-11

SHEET 3 OF 5

AMENDMENT

To: Bruce Preston
From: Dean James
Date: 9/4/84

SINGLE-FAILURE ANALYSIS FOR THE NEUTRON MONITORING AND PROCESS RADIATION MONITORING SYSTEMS

HOPE CREEK GENERATION STATION
PUBLIC SERVICE ELECTRIC AND GAS

AUGUST 1984

DBJ: rm/A08311*-1
8/31/84

SINGLE-FAILURE ANALYSIS FOR THE NEUTRON MONITORING AND PROCESS RADIATION MONITORING SYSTEMS

Some of the safety-related portions of the neutron monitoring system (NMS) and the process radiation monitoring system (PRMS) for the Hope Creek Generating Station (HCGS) are not designed and built to conform to the literal separation guidelines of Regulatory Guide 1.75. This analysis establishes the acceptability of these portions of the NMS and PRMS by demonstrating that they meet the single-failure criteria of IEEE Standard 279, which requires that the consequences of any single, design-basis failure event in a safety-related portion of the systems be tolerated without the loss of any safety function.

Portions of NMS and PRMS External to the NMS and PRMS Panels

See Figure 7.1-1 of the HCGS FSAR for the separations concept of the reactor protection system (RPS) and its relationship to the NMS.

Under the reactor vessel, cables from the individual, local power-range monitor (LPRM) detectors and from the individual intermediate-range monitor (IRM) detectors are grouped to correspond with the RPS trip channel designations. These cable groupings are run in conduit from the vessel pedestal area to the NMS and PRMS panels.

The radiation monitors on the main-steam lines are physically separated. The cabling from the individual sensors to the panels is run in separate metallic conduit.

Cabling from the NMS and PRMS panels to the RPS cabinets is also run in metallic conduit, providing electrical isolation and physical separation of the NMS and PRMS cabling associated with the RPS system.

It is concluded that the safety-related portions of the NMS and PRMS external to the NMS and PRMS panels adequately conform to the separation criteria of Regulatory Guide 1.75.

Single Failure in the NMS and PRMS Panels

Figures 1 and 2 depict schematically the physical arrangement of the equipment in NMS and PRMS panels H11-P608, H11-P635, and H11-P636. The designs of these panels are similar to those of NMS and PRMS panels used in several BWR plants accepted by the NRC.

The layouts of the panels and the assignments of specific RPS trip logic circuitry provides the designs with the required tolerance to postulated single failures. The worst-case single failure would be the loss of any combination of trip signals within one bay of any panel. However, the loss of any bay and its associated wiring would not prevent a scram. A valid scram signal would be transmitted via the other bays because of the redundancy in the panel designs and the interconnections to the RPS (see Figure 7.1-1 of the HCGS FSAR).

The eight IRM channels and the six average power range monitor (APRM) channels are electrically isolated and physically separated. Within the IRM and APRM modules, analog outputs are derived for use with control room meters, recorders, and the process computer. Electrical isolation at the interfaces would prevent any single failure from influencing the trip unit output.

Physical Separation in the NMS and PRMS Panels

Adequate separation in the NMS and PRMS panels is achieved by using the bay design depicted in Figures 1 and 2, by using relay coil-to-contact as sufficient separation/isolation, and by separation between divisions/channels/wiring. Where conformation with Regulatory Guide 1.75 separation criteria cannot be achieved, the best-effort design is used.

Circuits that provide inputs to different divisions of the RPS are physically separated by airgaps or by the walls between the bays. Within the panels, where the cable and wiring runs to the different RPS divisions do not conform to the Regulatory Guide 1.75 separation criteria, fire-resistant "Sil-Temp" tape is wrapped around the cables and wires. This eliminates the possibility of fault propagation between the RPS divisions. In accordance with paragraph 5.6.2 of IEEE Standard 384, this tape has been demonstrated to be acceptable.

Separated ducts are provided in the panel for the incoming circuit wires from the sensors that belong to UPS Bus 1 or Bus 2.

As shown in Figure 3, the isolation/separation precludes the propagation from outside the NMS cabinets of failures that could cause the loss of any safety function.

NMS/PRMS Interface to RPS

Although the LPRM sensors are not required to meet Class 1E requirements, the design bases of the APRMs specify that the LPRM signals used for the APRMs be so selected, powered, and routed that the APRMs do meet applicable safety criteria. The LPRM signal conditioners and associated power supplies are isolated and separated into groups.

The logic circuitry for the NMS and PRMS scram trip signals conforms to the single-failure criteria. The contact configurations and failure consequences associated with IRM A (see Figure 4) and APRM A (see Figure 5) are typical of the other trip channels and are described in what follows.

- With the reactor scram mode switch in the "Shutdown," "Refuel," or "Startup" positions, IRM A upscale or inoperating signals (unless bypassed) or APRM A upscale or inoperative signals (unless bypassed) would produce a channel trip of the output relay.

- With the reactor system mode switch in the "Run" position, IRM A upscale or inoperative signals (unless bypassed) and an APRM A downscale signal (unless bypassed) or APRM A upscale neutron trip or upscale thermal trip or inoperative signals (unless bypassed) would produce a channel trip of the output relay.

- A trip of the channel output relay for IRM A and APRM A or a trip of the channel output relay for IRM E and APRM E would produce an RPS A1 channel trip. In PRMS, the log radiation monitor A would produce an RPS A1 channel trip (see Figure 6).

For NMS, one tripped (unbypassed) channel on the RPS trip system would cause a half scram. 'f one APRM bay were to fail in an untripped condition, the remaining bays would be capable of sending RPS sufficient scram signals to produce a full scram, even if one of them were bypassed.

As shown in Figures 2 and 7, if one bay of panels H11-P635 or H11-P636 were to fail in an untripped condition, the remaining bays would be capable of sending sufficient RPS signals even if one of the IRM channels were bypassed. The IRM bypass switches can bypass one IRM channel at a time.

Similarly for PRMS, if one bay were to fail in an untripped condition, the remaining bays would be capable of sending sufficient RPS trip signals to produce a full scram.

Common Power Supply Justification

The NMS is supplied with 120-Vac, 60-Hz power from UPS busses 1 & 2. A design change has been authorized for the installation on each bus of redundant electrical protection assemblies (EPAs), which will monitor the incoming voltage and frequency.

Any fault in one NMS channel could not cause an unsafe failure in another channel sharing the same low voltage power supply because 10-amp fuses are installed for wire protection, and the power supplies are designed with over-voltage and over-current protection circuitry at their output.

The PRMS is supplied with 120-Vac, 60-Hz power from RPS busses A and B. EPAs are already installed on each bus to provide voltage and frequency protection.

Any fault in one PRMS channel could not cause an unsafe failure in another channel sharing the same power supply because 5-amp fuses are installed for wire protection, and the power supplies are designed with over-voltage and over-current protection circuitry at their output.

Because of the fail-safe NMS/PRMS logic configuration, a loss of one supply would result in a half scram signal to RPS. Loss of both supplies would result in a full scram.

Common Associated Circuit Interfaces

Nonessential (associated) circuits to common information equipment are current limited and protected such that their failure cannot jeopardize an adjacent circuit.

Figure 8 provides an example of an associated circuit interface on LPRM card Z11 At the zero-to-160-mV computer output, the card is protected with a 30-MA fuse. The zero-to-10-V output to the rod block monitor has an additional isolator protection for the card.

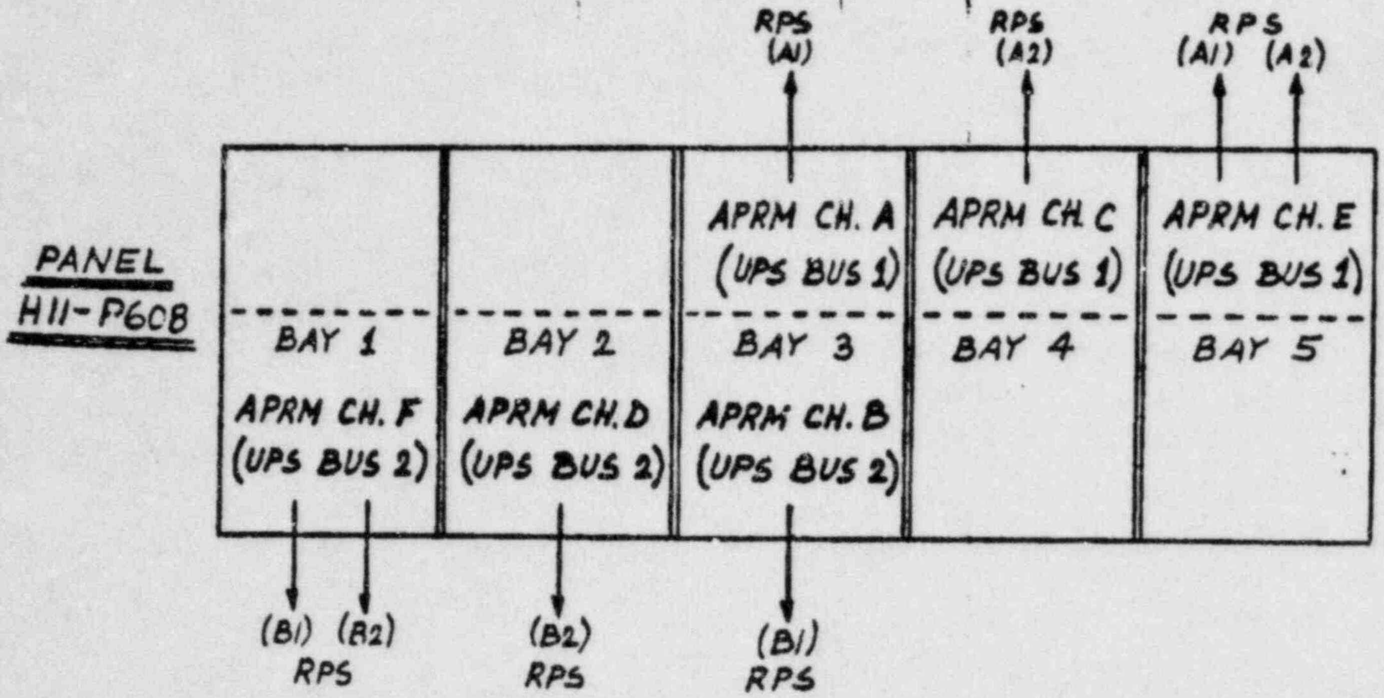


FIGURE 1 - APRM Panel Assignment

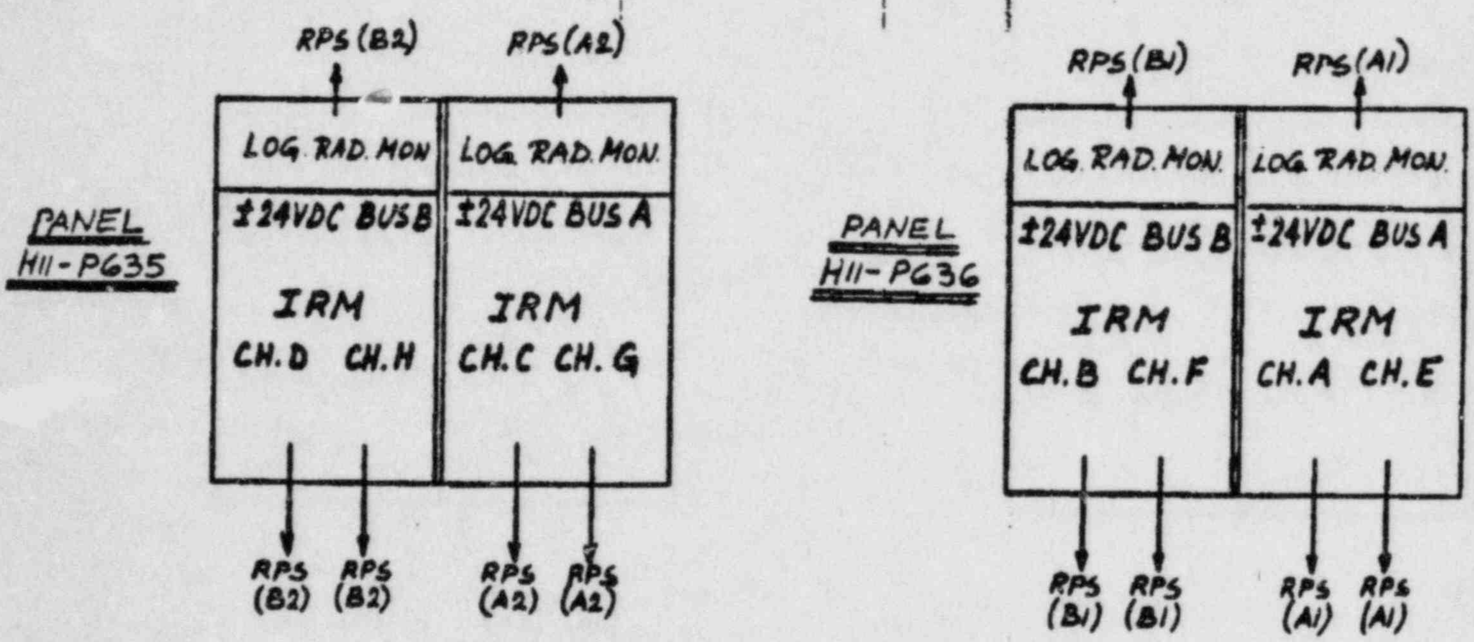


FIGURE 2 - Radiation Monitoring Panels

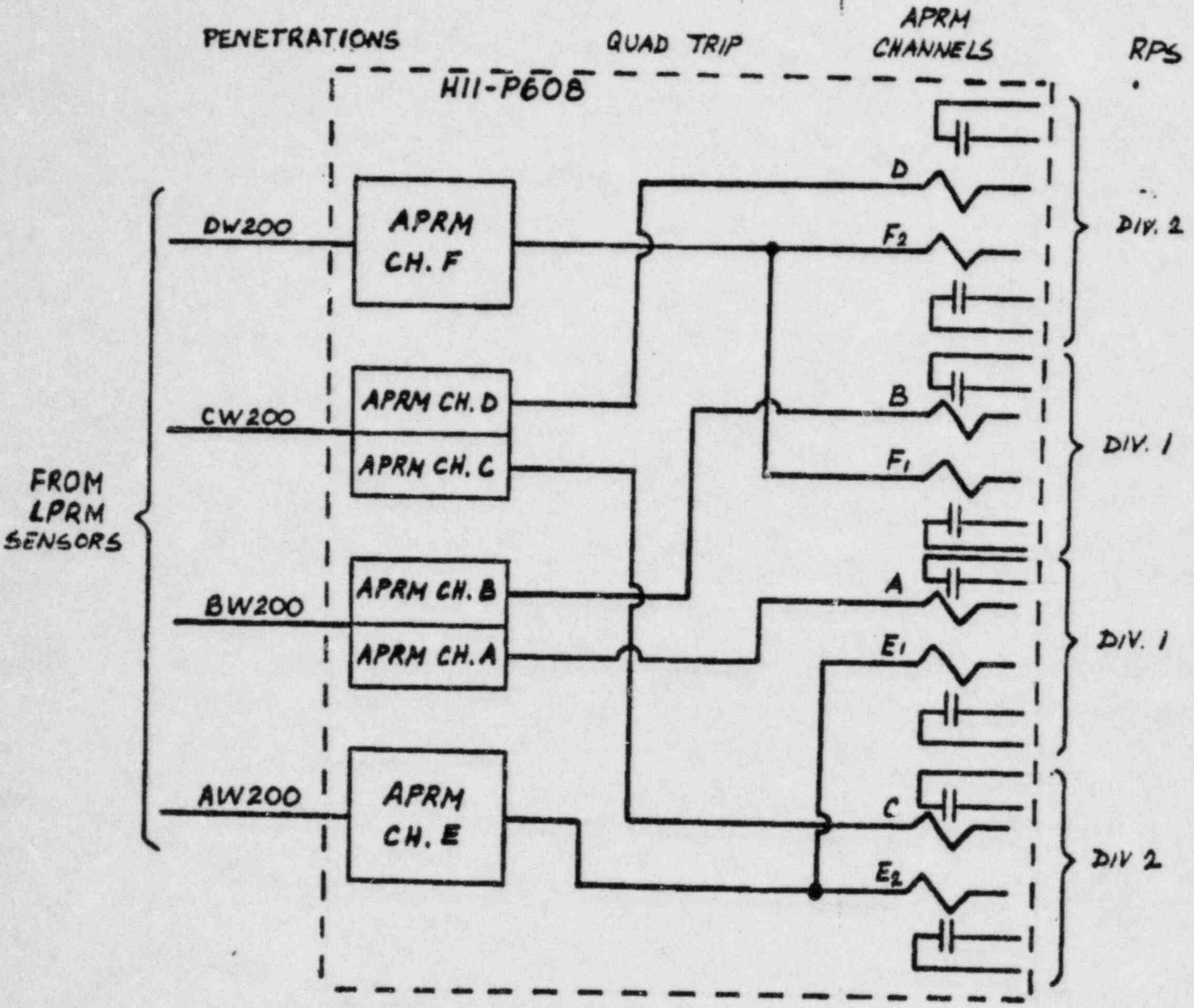


FIGURE 3 - APRM/RPS Signal Separation

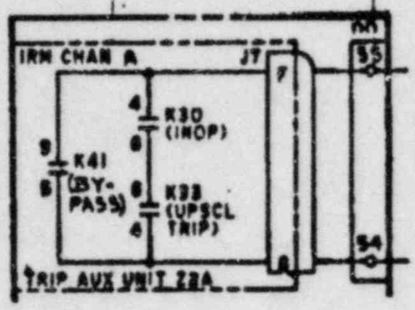


FIGURE 4 - IRM Channel A

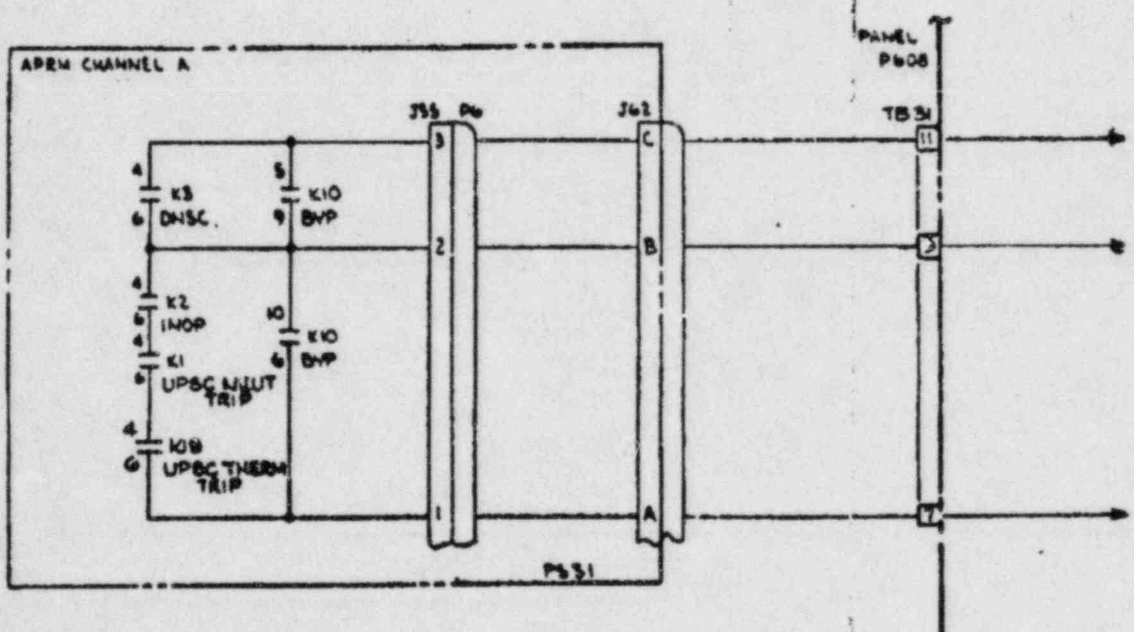


FIGURE 5 - APRM Channel A

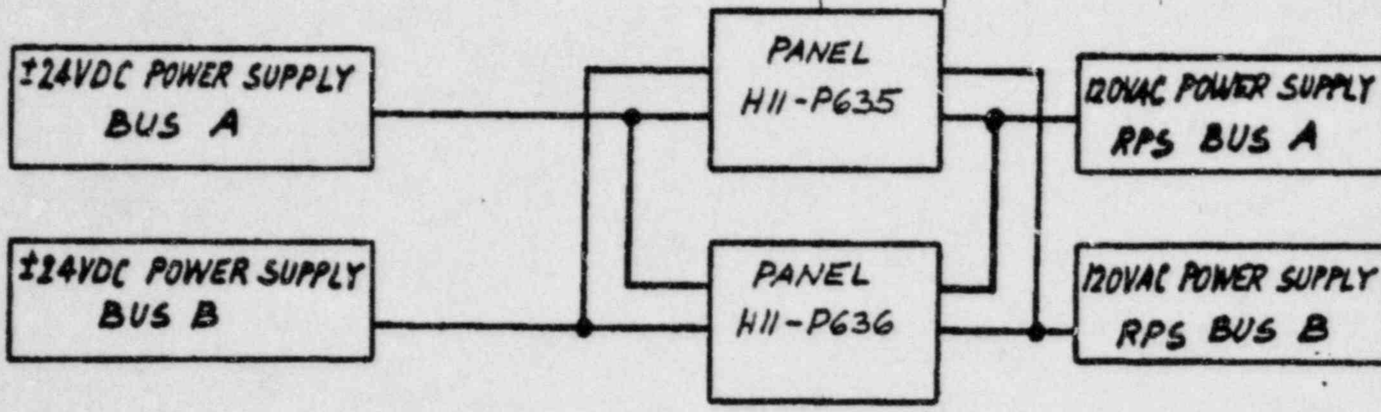


FIGURE 7 - Radiation Monitoring Panels Power Sources

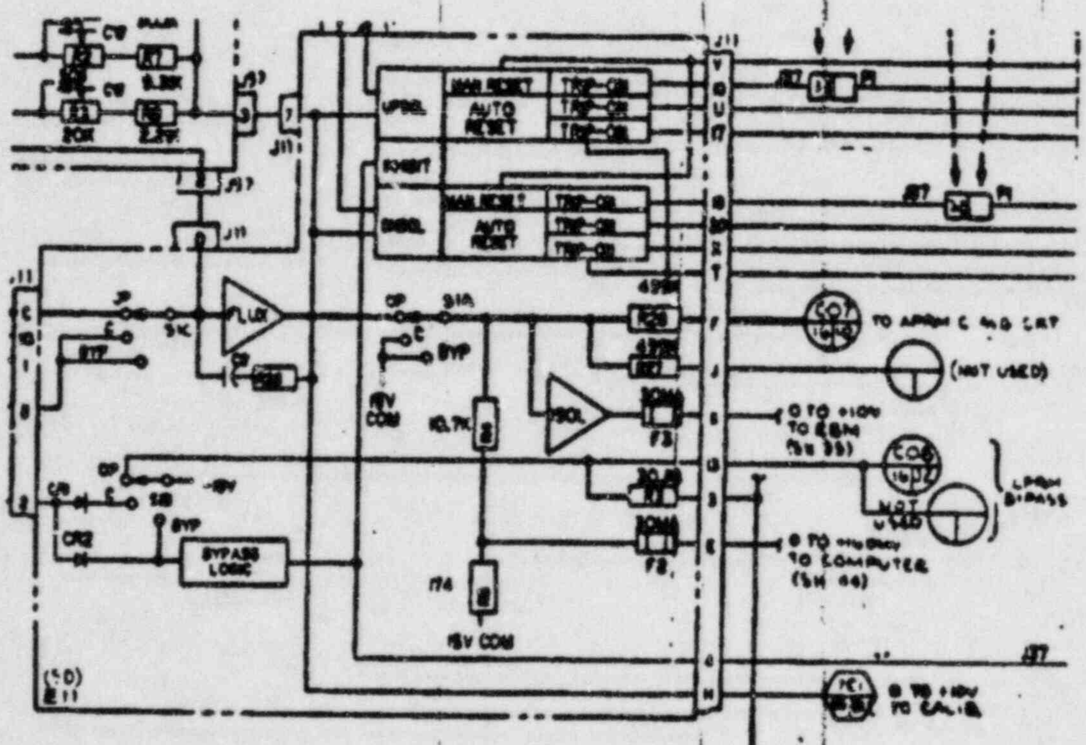


FIGURE 8 - LPRM Circuit Card

ATTACHMENT 5

Response

The HPCI and RCIC room unit coolers and their support systems are designed to withstand the consequences of a complete loss of offsite ac power since these are powered from onsite diesel generators. Each HPCI and RCIC room is provided with a 100%-capacity redundant unit cooler.

- II.K.3.25 EFFECT OF LOSS OF AC POWER ON PUMP SEALS

Position

The licensees should determine, on a plant-specific basis, by analysis or experiment, the consequences of a loss of cooling water to the reactor recirculation pump seal coolers. The pump seals should be designed to withstand a complete loss of alternating current power for at least 2 hours. Adequacy of the seal design should be demonstrated. The results of the evaluation and proposed modifications are due by July 1, 1981. Modifications are to be implemented by January 1, 1982.

Clarification

The intent of this position is to prevent excessive loss of reactor coolant system inventory following an anticipated operational occurrence. Loss of alternating current power for this case is construed to be loss of offsite power. If seal failure is the consequence of loss of cooling water to the reactor coolant pump seal coolers for 2 hours, due to loss of offsite power, one acceptable solution would be to supply emergency power to the component cooling water pump.

Response

At HCGS, cooling to the reactor recirculation pump seals is provided by the reactor auxiliaries cooling system (RACS). RACS is automatically energized from the Class 1E standby diesel generators during LOP.

PSE&G concurs with the BWROG study of this issue. BWROG submittals to the NRC on September 21, 1981, and September 2, 1982 provided test data showing very small seal leakage (on the order of 1 gpm) for a loss of seal cooling for longer than two hours. These results are applicable to the Byron-Jackson pumps used at HCGS. The normal or emergency controls for reactor water level could easily accommodate this small leakage rate.