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Washington Public Power Supply System
A JOINT OPERATING AGENCY

P. O. Box 968 3000 Geo. Washington Way Richland, Washington 99352 Phone (509) 375-5000

March 23, 1979
G02-79-50

Docket No. 50-397

Director, Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. S. A. Varga, Chief
Branch No. 4
Division of Project Management

Subject: WPPSS NUCLEAR PROJECT NO. 2
FSAR
AMENDMENT NO. 3

Dear Mr. Varga:

The Washington Public Power Supply System herewith submits sixty (60) copies of Amendment No. 3 to its Final Safety Analysis Report. Amendment 3 incorporates NRC questions up through the Acceptance Review into the FSAR plus other minor changes.

Pursuant to 10CFR 2.101, we will, within ten days of this filing, furnish to you an affidavit reflecting our distribution of this amendment to your designated distribution list.

Very truly yours,

D L Renberger

D. L. RENBERGER
Assistant Director
Technology

DLR:OKE:cph

cc: JJ Verderber - B&R, N.Y.
JJ Byrnes - B&R, N.Y.
HR Canter - B&R, N.Y.
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WPPSS Nuclear Project No. 2
FSAR Amendment No. 3

STATE OF WASHINGTON)
) ss
COUNTY OF BENTON)

D. L. RENBERGER, Being first duly sworn, deposes and says: That he is the Assistant Director, Technology, for the WASHINGTON PUBLIC POWER SUPPLY SYSTEM, the applicant herein; that he is authorized to submit the foregoing on behalf of said applicant; that he has read the foregoing and knows the contents thereof; and believes the same to be true to the best of his knowledge.

US NRC DISTRIBUTION SERVICES BRANCH
30 AM 8 25
DISTRIBUTION SERVICES UNIT

DATED March 14, 1979

D L Renberger
D. L. RENBERGER

On this day personally appeared before me D. L. RENBERGER to me known to be the individual who executed the foregoing instrument and acknowledged that he signed the same as his free act and deed for the uses and purposes therein mentioned.

GIVEN under my hand and seal this 14th day of March, 1979.

Reba B. Helgeson
Notary Public in and for the State of Washington
Residing at Richland

[illegible]

[illegible]

W/ ltr: 3/23/79
#7904300267

FINAL SAFETY ANALYSIS REPORT

CONTENTS

FINAL SAFETY ANALYSIS REPORT (Continued)

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010.007	1	1
010.008	1	1
010.009	1	1
010.010	1	21
010.011	9	9
010.012	2	5
010.013	2	5,25
010.014	3	5
010.015	1	13
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010.025	1	5
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010.035	1	21
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010.053	1	21
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022.009	1	1
022.010	1	13
022.011	1	1
022.012	1	3
022.013	1	3
022.014	1	13
022.015	1	3
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022.019	1	3
022.020	1	3
022.021	1	3
022.022	1	3
022.023	1	3
022.024	1	11
022.025	1	14
022.026	1	3
022.027	1	14
022.028	1	3
022.029	1	3
022.030	1	3
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022.032	2	5
022.033	1	5
022.034	1	5
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022.036	1	5
022.037	2	5
022.038	1	5

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022.041	1	5
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022.043	2	5,14
022.044	1	5
022.045	1	5
022.046	1	5
022.047	1	5
022.048	11	5,26
022.049	4	5
022.050	2	5
022.051	1	5
022.052	1	5
022.053	2	20
022.054	2	20
022.055	2	20
022.056	2	20
022.057	1	20
022.058	1	20
022.059	2	20

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
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022.061	5	20
022.062	1	20
022.063	3	20
022.064	1	21
022.065	1	21
022.066	1	21
022.067	1	21
022.068	1	21
022.069	1	21
022.070	1	21
022.071	1	21
022.072	1	21
022.073	1	21
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022.075	1	21
022.076	1	21
022.077	1	21
022.078	1	26
022.079	1	21
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022.082	1	21
022.083	1	21
022.084	1	21
022.085	1	21
022.086	1	21
022.087	1	21
022.088	1	21
022.089	1	21
022.090	1	21
022.091	1	21
022.092	1	21
022.093	3	21
022.094	2	21
022.095	1	21
022.096	1	21
022.097	1	21
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022.099	2	21
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022.107	8	20
022.109	2	20
022.110	1	20
022.111	1	20
022.112	2	20
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QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
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031.001(gg)	1	0
031.001(hh)	1	14
031.001(ii)	1	14
031.002	1	0
031.003	1	0
031.004	1	0
031.005	1	14
031.006	2	14
031.007	1	0
031.008	1	0
031.009	4	0
031.010	4	0,14
031.011	1	14
031.012	1	0
031.013	1	14
031.014	1	3
031.015	3	0,14
031.016	2	0,14
031.017	2	0
031.018	2	0

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
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031.020	1	0
031.021	2	14
031.022	1	14
031.023	1	25
031.024	1	0
031.025	2	0
031.026	1	10
031.027	1	14
031.028	1	14
031.029	1	0
031.030	2	0,14
031.031	1	14
031.032	1	0
031.033	1	0
031.034	2	0
031.035	1	14
031.036	1	14
031.037	1	14
031.038	2	14
031.039	2	0,14

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031.041	1	14
031.042	1	0
031.043	1	0
031.044	2	0,14
031.045	1	14
031.046	1	14
031.047	1	14
031.048	3	0,14
031.049	1	0
031.050	3	0
031.051	1	0
031.052	1	14
031.053	2	0
031.054	1	14
031.055	1	13
031.056	1	10
031.057	1	10
031.058	3	3,11
031.059	2	3,10
031.060	1	3

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
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031.062	1	3
031.063	1	3
031.064	1	3
031.065	1	3
031.066	2	3
031.067	1	3
031.068	1	3
031.069	1	14
031.070	5	3,21
031.071	1	3
031.072	1	14
031.073	1	3
031.074	1	3
031.075	1	14
031.076	3	5,14,25
031.077	1	3
031.078	2	3,14
031.079	2	3
031.080	7	10,13
031.081	2	17

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
031.082	1	10
031.083	1	10
031.084	1	10
031.085	1	10
031.086	1	10
031.087	5	10
031.088	1	10
031.089	1	10
031.090	1	10
031.091	1	10
031.092	2	10
031.093	1	10
031.094	1	10
031.095	1	10
031.096	1	10
031.097	1	10
031.098	1	10
031.099	1	10
031.100	1	23
031.101	1	10
031.102	1	10

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
031.103	3	10
031.104	2	10
031.105	2	10
031.106	1	10
031.107	1	10
031.108	2	10
031.109	1	10
031.110	1	10
031.111	1	10
031.112	1	10
031.113	1	10
031.114	2	21
031.115	1	21
031.116	1	21
031.117	3	23
031.118	1	23
031.119	1	21
031.120	1	21
031.121	1	21
031.122	1	21
031.123	1	21

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
031.124	2	21
031.125	1	21
031.126	2	21
031.127	1	21
031.128	1	23
031.129	1	21
031.130	1	21
031.131	2	21
031.132	1	21
031.133	1	21
031.134	1	21
040.001	2	0
040.002	1	0
040.003	1	0
040.004	1	1
040.005	1	0
040.006	1	0
040.007	1	0
040.008	1	0
040.009	1	14
040.010	2	0

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040.011	1	0
040.012	1	0
040.013	1	0
040.014	1	0
040.015	1	26
040.016	1	0
040.017	1	26
040.018	1	26
040.019	1	26
040.020	7	0
040.021	1	0
040.022	1	0
040.023	2	26
040.024	2	0
040.025	1	0
040.026	2	26
040.027	1	0
040.028	1	0
040.029	1	0
040.030	1	0
040.031	1	0

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040.033	1	0
040.034	23	26
040.035	1	26
040.036	2	7,25
040.037	1	7
040.038	1	7
040.039	4	7,11
040.040	1	7
040.041	1	7
040.042	1	7
040.043	1	14
040.044	2	7
040.045	6	26
040.046	1	26
040.047	3	26
040.048	1	7
040.049	1	7
040.050	1	26
040.051	1	7
040.052	1	26

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
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040.054	1	26
040.055	1	7
040.056	1	26
040.057	1	7
040.058	1	7
040.059	1	26
040.060	2	7,14
040.061	1	26
040.062	1	7
040.063	1	7
040.064	1	7
040.065	1	7
040.066	1	7
040.067	1	7
040.068	1	7
040.069	1	7
040.070	1	14
040.071	1	7
040.072	1	7
040.073	1	7

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040.076	4	20
040.077	3	20
040.078	4	20
040.079	10	20
040.080	2	25
040.081	2	26
040.082	3	21,26
040.083	1	26
040.084	4	26
040.085	1	26
040.086	1	26
040.087	1	21
040.088	2	23,26
040.089	1	21
110.001	1	9
110.002	1	25
110.003	1	9
110.004	1	25
110.005	1	9

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110.007	1	9
110.008	1	9
110.009	1	9
110.010	1	9
110.011	1	9
110.012	1	9
110.013	1	9
110.014	1	9
110.015	1	9
110.016	1	9
110.017	1	25
110.018	1	9
110.019	4	9
110.020	1	9
110.021	1	9
110.022	1	25
110.023	1	9
110.024	1	9
110.025	1	9
110.026	1	9

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110.028	2	9
110.029	1	9
110.030	2	9
110.031	1	9
110.032	1	9
110.033	1	9
110.034	1	21
110.035	1	9
110.036	2	9
110.037	2	9
121.001	3	5
121.002	8	5
121.003	1	5
121.004	1	5
121.005	1	5
121.006	1	5
121.007	1	13
121.008	10	5,10
121.009	1	5
121.010	2	7

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
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121.012	1	23
121.013	1	23
121.014	1	23
121.015	2	23
121.016	1	23
121.017	1	23
121.018	1	23
121.019	1	23
130.001	1	1
130.002	1	1
130.003	1	1
130.004	1	1
130.005	1	1
130.006	1	1
130.007	1	1
130.008	1	1
130.009	1	13
130.010	1	8
130.011	1	8
130.012	1	8

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130.014	1	8
130.015	1	8
130.016	1	8
130.017	1	8
130.018	1	8
130.019	1	8
130.020	1	23
130.021	1	8
130.022	1	8
130.023	1	8
130.024	5	23
130.025	1	8
130.026	1	8
130.027	1	8
130.028	1	8
130.029	1	8
130.030	1	8
130.031	1	8
130.032	1	8
130.033	1	8

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
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130.035	4	8,25
130.036	1	8
130.037	1	8
130.038	2	8
130.039	7	8
130.040	1	8
130.041	1	8
130.042	8	8
130.043	1	8
130.044	1	8
130.045	3	12
130.046	3	12
130.047	1	12
130.048	2	12,23
130.049	3	12
130.050	1	23
130.051	2	23
130.052	1	23
130.053	4	23
130.054	1	23

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
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130.056	2	23
130.057	1	23
130.058	1	23
130.059	1	23
130.060	4	23
130.061	2	23
130.062	3	23
130.063	1	23
130.064	2	23
130.065	1	23
130.066	2	23
130.067	2	23
130.068	3	23
130.069	2	23
130.070	2	23
130.071	1	23
130.072	2	23
130.073	2	23
130.074	1	23
130.075	1	23

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
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130.077	1	23
130.078	1	23
210.001	1	0
211.002	1	8
211.003	1	8
211.004	1	14
211.005	1	8
211.006	1	21
211.007	1	14
211.008	2	8,14
211.009	1	8
211.010	2	21,23
211.011	1	8
211.012	2	8,23
211.013	1	8
211.014	2	8
211.015	1	8
211.016	3	8
211.017	2	8
211.018	1	21

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
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211.020	1	8
211.021	2	8
211.022	1	8
211.023	1	8
211.024	1	8
211.025	2	23
211.026	1	8
211.027	4	8
211.028	2	8
211.029	1	8
211.030	1	8
211.031	6	23
211.032	1	21
211.033	4	8
211.034	1	8
211.035	1	8
211.036	1	8
211.037	1	17
211.038	3	8
211.039	2	8

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
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211.041	1	8
211.042	1	8
211.043	1	8
211.044	1	8
211.045	1	8
211.046	1	8
211.047	1	8
211.048	2	8
211.049	1	23
211.050	1	11
211.051	15	11,23
211.052	1	11
211.053	1	11
211.054	1	23
211.055	1	11
211.056	2	11
211.057	1	11
211.058	3	11
211.059	1	11
211.060	1	11

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
211.061	2	11
211.062	1	11
211.063	1	11
211.064	1	11
211.065	2	11
211.066	2	11
211.067	1	11
211.068	1	11
211.069	1	11
211.070	1	11
211.071	1	11
211.072	3	23
211.073	1	11
211.074	1	11
211.075	1	11
211.076	2	11
211.077	4	11
211.078	1	11
211.079	4	11
211.080	2	11
211.081	2	11

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
211.082	3	11
211.083	1	11
211.084	1	23
211.085	4	11
211.086	1	11
211.087	5	21
211.088	2	11
211.089	5	11
211.090	1	11
211.091	1	11
211.092	3	23
211.093	1	11
211.094	1	11
211.095	1	11
211.096	1	20
211.097	1	11
211.098	1	11
211.099	1	23
211.100	1	11
211.101	2	11
211.102	1	11

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
211.103	1	11
211.104	1	11
211.105	3	11
211.106	1	11
211.107	1	21
211.108	1	21
211.109	1	21
211.110	1	21
211.111	1	21
211.112	1	21
211.113	2	21
211.114	1	21
211.115	1	20
211.116	3	21
211.117	1	21
211.118	1	21
211.119	1	21
211.120	2	21
211.121	1	21
211.122	1	21
211.123	1	21

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
211.124	1	20
211.125	1	21
211.126	1	21
211.127	2	20
211.128	2	21
211.129	1	23
211.130	1	21
211.131	1	21
211.132	1	21
211.133	1	21
211.134	1	21
211.135	1	21
211.136	1	23
211.137	1	20
211.138	1	21
211.139	2	20
211.140	1	20
211.141	1	20
211.142	1	20
211.143	2	21
211.144	2	21

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
211.145	1	20
211.146	1	23
211.147	1	20
211.148	3	23
211.149	1	20
211.150	1	21
211.151	1	20
211.152	1	21
211.153	2	20
211.154	1	20
211.155	2	20
211.156	2	20
211.157	2	20
211.158	1	20
211.159	1	23
211.160	1	20
211.161	1	20
211.162	1	20
211.163	1	20
211.164	2	21
211.165	2	20

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
211.166	2	20
211.167	1	20
211.168	1	21
211.169	2	20
211.170	1	20
211.171	4	21
211.172	1	20
211.173	1	21
211.174	1	21
211.175	1	20
211.176	1	20
211.177	1	20
211.178	1	20
211.179	1	20
211.180	1	21
211.181	1	20
211.182	1	21
211.183	1	20
211.184	1	20
211.185	1	21
211.186	1	20

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
211.187	1	20
211.188	1	20
211.189	2	21
211.190	1	20
211.191	1	20
211.192	1	20
211.193	1	20
211.194	1	20
211.195	2	20
211.196	1	21
211.197	1	23
211.198	2	21
211.199	2	21
211.200	1	20
211.201	1	21
211.202	2	21
211.203	1	21
211.204	1	21
211.205	2	20
211.206	11	21
211.207	2	20,21

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
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211.209	1	23
211.210	1	20
211.211	1	21
211.212	1	21
211.213	1	21
212.001	1	3
212.002	1	3
212.003	2	5,14
212.004	1	3
221.001	2	7
221.002	1	7
221.003	1	7
221.004	1	7
221.005	1	7
221.006	1	7
221.007	1	7
221.008	1	7
221.009	1	7
221.010	2	7
221.011	1	7

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221.013	1	7
222.001	1	8
222.002	2	8,25
222.003	1	8
222.004	1	8
231.001	1	3
231.002	3	3
231.003	1	5
231.004	2	20
231.005	7	20
232.001	1	3
232.002	1	5
232.003	1	5
232.004	1	5
232.005	2	5
271.001	3	21
271.002	269	21
271.003	3	21
271.006	1	21
281.001	2	21

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
281.002	1	21
281.003	1	21
281.004	1	21
281.005	1	21
281.006	1	21
281.007	1	21
281.008	1	21
281.009	2	21
281.010	1	21
281.011	1	21
281.012	1	21
281.013	1	21
281.014	1	21
311.001	1	23
312.001	1	1
312.002	1	1
312.003	1	1
312.004	1	1
312.005	1	1
312.006	1	1
312.007	1	1

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
312.009	1	8
312.010	1	1
312.011	1	1
312.012	2	1
312.013	1	1
312.014	1	1
312.015	1	8
312.016	1	5
312.017	1	14
312.018	1	11
312.019	1	5
321.001	1	14
321.002	1	1
321.003	1	13
321.004	1	21
321.005	4	5,20
331.001	1	13
331.002	2	13
331.003	2	1
331.004	1	1
331.005	1	1

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
331.006	1	1
331.007	1	1
331.008	1	1
331.009	1	1
331.010	1	1
331.011	1	1
331.012	1	1
331.013	1	1
331.014	1	1
331.015	2	5
331.016	1	5
331.017	1	5
331.018	1	5
331.019	1	5
331.020	1	5
331.021	1	13
331.022	1	5
331.023	1	5
331.024	1	5
331.025	2	20
331.026	2	20

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
331.027	1	20
331.028	2	20
331.029	1	20
360.001	1	1
360.002	1	1
360.003	1	1
360.004	3	10
360.005	6	10
360.006	2	21
360.007	3	21
360.008	4	21
360.009	7	21
360.010	3	21
360.011	2	21
360.012	2	21
360.013	3	21
360.014	13	23
360.015	7	23
360.016	3	23
360.017	10	23
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360.020	10	23
360.021	1	23
360.022	3	23
361.001	1	21
361.002	2	21
361.003	1	21
361.004	28	21
361.005	6	21
361.006	1	21
361.007	2	21
361.008	2	21
361.009	3	21
361.010	3	21
361.011	2	21
361.012	1	21
361.013	2	21
361.014	1	21
361.015	3	23
361.016	4	23
361.017	29	23

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361.019	37	23
362.001	1	3
362.002	2	3
362.003	1	3
362.004	2	3
362.005	1	5
362.006	1	5
362.007	1	5
362.008	1	5
362.009	2	5
362.010	5	21
362.011	6	26
362.012	2	21
362.013	1	21
362.014	3	21
371.001	1	23
371.002	1	1
371.003	1	1
371.004	1	1
371.005	1	1

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
371.006	1	5
371.008	1	23
371.009	1	5
371.010	1	5
371.011	1	5
371.012	1	5
371.013	1	5
371.014	1	5
371.015	5	23
371.016	3	23
371.017	2	23
371.018	3	23
371.019	2	23
371.020	1	23
371.03E	1	20
371.04E	1	20
371.05E	1	20
371.06E	1	20
371.07E	1	20
372.001	1	1
372.002	1	1

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
372.003	1	1
372.004	1	1
372.005	1	1
372.006	1	1
372.007	1	3
372.008	14	5
372.009	1	5
372.010	1	5
372.011	1	5
372.012	1	5
372.013	1	5
372.014	7	5
372.015	4	5
372.016	5	5
372.017	1	5
421.001	1	20
421.002	1	20
421.003	1	20
421.004	2	20
421.005	1	20
421.006	1	20

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
421.007	1	20
421.008	1	21
421.009	1	20
421.010	1	20
421.011	1	20
421.012	1	20
421.013	1	20
421.014	1	20
421.015	1	20
421.016	1	20
421.017	1	20
421.018	1	20
421.019	1	20
421.020	2	20
421.021	1	20
421.022	1	20
421.023	1	20
421.024	1	20
421.025	1	20
421.026	1	20
421.027	1	20

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
421.028	1	20
421.029	1	20
421.030	1	20
421.031	1	20
421.032	1	20
421.033	1	20
421.034	1	20
421.035	1	20
421.036	1	20
421.037	1	20
421.038	1	20
421.039	1	20
421.040	1	20
421.041	2	21
421.042	1	20
421.043	17	20,25
422.001	2	7
422.002	1	26
422.003	1	26
422.004	3	26
422.005	1	7

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422.006	3	7
422.007	2	7
422.008	2	7
422.009	1	7
423.001	1	1
423.002	1	1
423.003	1	1
423.004	1	0
423.005	1	1
423.006	2	1
423.007	1	1
423.008	1	1
423.009	1	1
423.010	1	1
423.011	4	7
423.012	1	7
423.013	1	7
423.014	1	7
423.015	1	7
423.016	2	7
423.017	1	7

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423.019	7	7
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423.021	3	26
423.022	1	7
423.023	12	7,26
423.024	1	7
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423.026	1	7
423.027	1	7
423.028	1	7
423.029	1	7
423.030	21	26
423.031	1	20
423.032	1	20
423.033	1	26
423.034	1	20
423.035	1	20
423.036	1	20
423.037	1	20
423.038	1	20

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
423.039	1	26
423.040	1	20
423.041	9	26
423.042	4	26
432.001	7	5
432.002	2	5
432.003	2	5
432.004	3	5
432.005	1	5
432.006	1	5
432.007	1	5
432.008	1	5
432.009	1	5
432.010	1	5
432.011	2	5
432.012	1	5
432.013	2	5
432.014	1	5
432.015	1	5
432.016	1	5
432.017	2	21

QUESTIONS	NRC QUESTIONS NO. OF PAGES	AMENDMENT
432.018	1	21
432.019	1	21
432.020	1	21
432.021	1	21
432.022	1	21
432.023	1	21
432.024	1	21
432.025	1	21
441.001	3	7
441.002	1	7
441.003	1	7
441.004	1	7
441.005	1	7
600.001	4	23
600.002	3	23
600.003	2	23
600.004	2	23
600.005	1	23
600.005	1	23
600.007	1	23
600.008	1	23
600.009	1	23

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031.078
031.079
031.080(m,o,p)

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031.001(z)
031.001(aa)
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022.068
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031.001(cc)
031.001(dd)
031.001(ff)
031.001(ii)
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031.080(n,q,r,s,t,u)
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Section 7.6 (Continued)

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031.032
031.054
031.080(m)
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331.026

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331.027

Section 12.2

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331.017

Section 12.3

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421.025
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421.027
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421.035
421.036
421.037
421.038
421.039
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421.041
421.042
421.043

APPENDIX H

031.115

QUESTIONS REFERENCING DAR

022.028	022.059	022.086
022.029	022.060	022.087
022.030	022.061	022.107
022.032	022.062	022.109
022.053	022.063	022.110
022.054	022.065	022.111
022.055	022.066	022.112
022.056	022.083	022.113
022.057	022.084	022.114
022.058	022.085	110.026
		110.027
		362.014

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022.034

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040.077
040.078
040.079
422.001
422.002

WNP-2 RESPONSE TO TMI

281.014

QUESTIONS REFERENCING QUESTIONS

Question	Question Referenced
010.056	010.021
010.062	010.029
010.065	010.034
022.067	022.020
022.069	022.018
	031.070
022.070	022.006
	022.035
022.078	022.048
022.079	022.049
022.080	022.050
031.058	031.001
031.059	031.001
031.060	031.001
031.061	031.039
031.062	031.001
	031.002
	031.023
031.063	031.006
	031.014
	031.016
	031.026
	031.047
031.064	031.009
031.065	031.010
031.066	031.009
	031.018
031.067	031.025
031.068	031.032
031.077	031.021
	031.037
031.078	031.050
031.083	031.006
	031.056
	031.059
031.090	031.032
031.094	031.033
031.097	031.026
031.099	031.030
040.035	040.034
130.050	220.001
130.051	220.002
130.052	220.003
130.053	220.004
130.054	220.005
130.055	220.006

QUESTIONS REFERENCING QUESTIONS (Continued)

Question	Question Referenced
130.056	220.007
130.057	220.008
130.058	220.009
130.059	220.010
130.060	220.011
130.061	220.012
130.062	220.013
130.063	220.014
130.064	220.015
130.065	220.016
130.066	220.017
130.067	220.018
130.068	220.019
130.069	220.020
130.070	220.021
130.071	220.022
130.072	220.023
130.073	220.024
130.074	220.025
130.075	220.026
130.076	220.027
130.077	220.028
130.078	220.029
211.006	010.049
211.107	010.037
211.108	010.038
211.109	010.039
211.122	010.050
211.123	010.051
211.125	010.052
211.126	010.053
211.130	010.044
211.131	010.045
211.133	010.046
211.134	010.047
211.135	010.048
211.211	211.185
211.212	212.003
312.016	022.048
371.001	371.016
371.008	371.016
371.016	371.001
	371.008
423.011	423.002
423.024	423.006
	423.007

QUESTIONS REFERENCING QUESTIONS (Continued)

Question	Question Referenced
	423.023
432.018	432.017
432.019	432.017
432.020	432.017
432.021	432.017
432.022	432.017
432.023	432.017
432.024	432.017
432.025	432.017

QUESTIONS WITH NO REFERENCES

005.001
010.059
010.060
010.061
010.063
010.064
022.031
022.036
022.039
022.050
022.051
022.052
022.064
022.071
022.097
022.098
022.099
022.100
022.101
022.102
022.103
022.104
022.105
022.106
031.001(e)
031.001(x)
031.001(gg)
031.042
031.122
040.026
040.034
040.039
040.040
040.041
040.042
040.044
121.011
121.012
121.013
121.014
121.015
121.016
121.017
121.018
121.019
210.001
211.213
212.002
231.005
232.005
271.001
271.002
271.003
271.006
312.005
312.015
331.002
331.015
371.015
371.018
371.019
371.020
372.002
372.016
423.029
423.016
432.017
600.001
600.002
600.003
600.004
600.005
600.006
600.007
600.008
600.009

QUESTIONS REFERENCING QUESTIONS (Continued)

Question	Question Referenced
432.018	432.017
432.019	432.017
432.020	432.017
432.021	432.017
432.022	432.017
432.023	432.017
432.024	432.017
432.025	432.017

QUESTIONS WITH NO REFERENCES

005.001
010.059
010.060
010.061
010.063
010.064
022.031
022.036
022.039
022.050
022.051
022.052
022.064
022.071
022.097
022.098
022.099
022.100
022.101
022.102
022.103
022.104
022.105
022.106
031.001(e)
031.001(x)
031.001(gg)
031.042
031.122
040.026
040.034
040.035
040.039
040.040
040.041
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040.044
121.011
121.012
121.013
121.014
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121.016
121.017
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210.001
211.213
212.002
231.005
232.005
271.001
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312.005
312.015
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331.015
371.015
371.018
371.019
371.020
372.002
372.016
423.029
423.016
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600.001
600.002
600.003
600.004
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Q. 005.001

In order that we might evaluate your compliance with the Codes and Standards Rule, Section 50.55a of 10 CFR Part 50, identify the edition Section III of the ASME Boiler and Pressure Vessel Code and the applicable code addenda for the following Quality Group A components within the reactor coolant pressure boundary identified in Table 3.2-1 of the FSAR. These components are: (1) the reactor pressure vessel; (2) the main steam and feedwater piping inboard of the outermost isolation valves; (3) the piping of other interconnecting systems inboard of the outermost isolation valves; (4) the main steam isolation valves; (5) the explosive valves of the standby liquid control systems; and (6) the system or isolation valves of other interconnecting systems.

Response:

Please see Table 005.001-1.

TABLE 005.001-2

Requested Item Per Question	WNP-2 Installation	Applicable Code Per 10CFR 50.55(a)*	Purchase Order Date	Actual Code Used
1. Reactor Pressure Vessel	N/A	1971 Code, no Addenda	11/71	1971 Code, Summer '71 Addenda (para. NB 3338.2 is Winter '71 Addenda)
2. Main Steam & Feedwater Piping	Shop Fabricated	1971 Code, Winter '73 Addenda	10/74	1971 Code, Winter '73 Addenda
	Field Fabricated	1971 Code, Summer '73 Addenda	5/74	1971 Code, Winter '73 Addenda***
	Main Steam	1971 Code, Winter '71 Addenda	9/72	1971 Code, Summer '72 Addenda
3. Piping of Inter-connecting Systems	Shop Fabricated	Same as (2)		
	Field Fabricated	Same as (2)		
	Recirc System Piping**			
4. MSIVs	N/A	1971 Code, Summer '71 Addenda	4/71	1971 Code, Winter '71 Addenda
5. SLCS Explosive Valves	N/A	1971 Code, Winter '72 Addenda	12/73	1971 Code, Winter '72 Addenda
6. Other Valves of Interconnecting Systems	Velan Valves I	1971 Code, Summer '73 Addenda	1/74	1971 Code, Winter '73 Addenda
	Anchor Darling Valves	1971 Code, Winter '72 Addenda	7/73	1971 Code, Winter '73 Addenda
	Velan Valves II	1971 Code, Summer '72 Addenda	1/73	1971 Code, Winter '72 Addenda
	Borg-Warner Valves	1971 Code, Summer '73 Addenda	5/74	1971 Code, Winter '73 Addenda
	Recirc and Crosby Valves**			

* CP issued 3/73

** See Table 5.2-5

*** The 1975 Summer Addenda, 1974 edition, paragraph NB-3610, is used for the design of 1-inch and smaller piping.

Q 010.1

Provide a tabulation of all safety-related components which are located outdoors and describe the protection provided for these components to prevent their being damaged by tornado-generated missiles, turbine missiles, or a seismic event. Include in this tabulation, all heating, ventilation and air conditioning system air intakes and exhausts. Identify the locations of the safety-related components, air intakes and exhausts on the plant arrangement drawings.

Response:

Structures which house safety-related components, and which are designed to resist tornado-generated missiles are discussed in 3.5.1.4.1. Safety-related components which must be protected from tornado-generated missiles are discussed in 3.5.2. Table 3.5-6 provides a tabulation of HVAC air intakes and exhausts, and refers to FSAR figures showing their locations.

Q. 010.002

Expanded 3.6.1 to include piping layout drawings for areas containing high and moderate energy lines whose failure can affect the performance of safety-related equipment. Provide a detailed analysis that demonstrates the method used to protect the reactor heat removal system from the effects of postulated piping system failures. Identify the assumptions and parameters used in your analysis such as flow rates through postulated cracks, pump room areas, sump capacities and floor drainage system capacities.

Response:

Figures 3.6-38 through 3.6-40 and 3.6-43 through 3.6-62 show piping configuration in areas where safety-related equipment could be affected by pipe rupture. Section 3.6.1.11.4 discusses the method of analysis used in evaluating the effect of pipe rupture on the ability to safely shut down the plant.

The approach used consists of evaluating the consequences of a postulated pipe break on any safety-related system, equipment, or components that could be affected, based on examination of the as-built conditions of the plant. The effects of each postulated pipe break are examined on a case-by-case basis. A single active component failure is then assumed in a system not affected by the pipe rupture, and the ability to bring the plant to a safe shutdown condition is evaluated. If safe shutdown can be attained, no protection is provided for the systems, equipment, and components damaged by the postulated rupture.

In the analysis for pipe breaks in moderate energy systems, flow rates through postulated cracks, pump room areas, sump capacities, and floor drainage system capacities are considered. Credit is taken for flow restrictions and system resistance. All components and equipment that could be sprayed or wetted are considered lost, unless contained in a water tight compartment. Also refer to 3.6.1.17 for a description of the capability to detect leakage and resultant flooding from fluid system piping ruptures.

A final walkdown of essential systems, equipment, and components has been performed to verify protection from the effects of postulated pipe ruptures.

Q 010.3

Provide a tabulation of all valves in the reactor pressure boundary and in other seismic Category I systems, as defined in Regulatory Guide 1.29, whose operation is relied upon either to assure safe plant shutdown or to mitigate the consequences of an accident (e.g., safety valves, stop valves, relief valves, stop-check valves, and control valves). The tabulation should identify the system in which it is installed, the type and size of valves, the actuation type(s), and the environmental conditions for which the valves are qualified.

Response:

A tabulation of safety related valves in the reactor pressure boundary and other Seismic Category I systems is provided in the IST program plan. (Tables 3.2-2 and 3.2-3 explain that ASME III Class 1, 2 valves and all safety-related Class 3 valves are Seismic Category I.

The IST program identifies the system in which each of these Seismic I valves are installed, the valve actuator, valve number, the type of valve and its ASME classification.

Valve sizes are identified on the associated system P&ID, supplied in 3.2 of the FSAR, since all line sizes are identified.

The valve actuator is also given on the P&ID.

The valve environmental qualification conditions for all safety-related Seismic I, components inside containment, outside containment and inside the drywell (those in the reactor pressure boundary and in other Seismic I systems) are given in 3.11 of the FSAR.

Q 10.4

Provide the results of an analysis of a postulated cask drop to demonstrate that this postulated event will not cause damage to spent fuel or any safety-related system or component which may be located under the travel path of the cask. Provide drawings that show the pathway of the cask in the vicinity of the spent fuel pool.

Response:

Please see 9.1.4.1, 9.1.4.2.10.3, and 9.1.4.3 for requested information. Figure 9.1-17 shows the cask travel path.

Q 010.5

Compare the outdoor humidity and temperatures that were used as the basis for the design of safety-related heating, ventilating and air conditioning (HVAC) systems to the extreme temperatures and humidity that may occur at the site. Provide justification for any differences between these extreme conditions and the design basis conditions of the HVAC systems.

Response:

This question has been addressed in 9.4 (page 9.4-2).

Q. 010.006

Provide the design basis, description, safety evaluation, testing and inspection requirements and drawings of the spent fuel pool area ventilating system.

Response:

During normal operating conditions ventilation of the spent fuel pool is provided by the Reactor Building ventilation system. The design basis, description, safety evaluation and testing of the spent fuel pool ventilating system is addressed in 9.4.2. The system is not classified as an Engineered Safety Feature.

The Reactor Building is isolated and the ventilation system fans are stopped during emergency conditions indicated by reactor vessel low water level, high drywell pressure and high radiation levels in the reactor building ventilation system exhaust (FAZ). During the emergency condition any vapor boil off from the spent fuel pool is processed through the Standby Gas Treatment System, as described in 6.5.

010.007

9.5.1 does not contain sufficient information to permit the staff to make a determination that the fire protection system design conforms with the guidelines in Appendix A of Branch Technical Position (BTP) APCSB 9.5.1. Provide a comparison between the criteria used in the design of your proposed fire protection system and the guidelines given in Appendix A of the cited BTP and justify any differences. Provide a fire hazards analysis for the WNP-2 facility and a description of the method of fire protection used for the power generation control console.

Response:

This information has been furnished in Appendix F, Fire Protection Evaluation.

For the Control Room Power Generation Control Complex cabinet fire protection, see the GF Topical Report NEDO-10466 Rev. 1 which was issued 10/1/77.

Q 010.8

Provide additional description of the main steam line isolation valves (MSIV's), including the type of valve, operator type, method of control, failure mode in the event of loss of compressed air, and other appropriate data. This information should demonstrate that the operation of the MSIV's for safe plant shutdown is supported by a safety-grade supply of compressed air and electrical power.

Response:

Please see 10.3.2, pages 10.3-2 and -3 for the requested information.

Q 010.9

Expand 10.4.5 to provide a detailed evaluation of the effects of a postulated failure of the circulating water system inside the turbine building, including a discussion of the following considerations:

a. The capability of detecting a failure in the circulating water system barrier (e.g., the rubber expansion joints), including a description of the method of detecting such a failure. Identify the design and operating pressures of the various portions of the circulating water system barrier and their relation to the pressures which could exist during malfunctions and failures in the system (e.g., rapid valve closure).

b. The time required to stop the circulating water flow (time zero being the instant failure) including all inherent delays such as operator reaction time, drop out times of the control circuitry and coastdown time.

c. For each postulated failure in the circulating water transport system barrier, provide the rate of rise of water in the associated spaces and the total height of the water when the circulating water flow either has been stopped or overflows to the site grade.

d. For each potentially flooded space, provide a discussion and drawing of the protective barrier provided for all safety-related systems that could be affected in the event of flooding. Include in your discussion the consideration given to passageways, pipe chases, and/or the cableways connecting the flooded spaces to the spaces containing safety-related systems or components outside the turbine building. Discuss the effect of the flood water on all potentially submerged safety-related electrical systems and components.

Response:

Please see expanded 10.4.5.3 for the requested evaluation.

Q. 010.010
(3.4.1)

Demonstrate that all piping and electrical penetrations in safety-related structures that are below the level of the Probable Maximum Flood are watertight.

Response:

As stated in 3.4.1.4.1 the plant site grade is higher than the design basis flood elevation resulting from the probable maximum precipitation (PMP) event. Due to the short duration of the PMP flood, the ground water level at the plant site is not affected. As stated in 3.4.1.4.2, piping and electrical penetrations are above the design basis groundwater level and therefore sealing against groundwater pressure is not required.

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Q. 010.011
(3.5)

We require you to provide an evaluation of the environmental effects resulting from a postulated failure of the main steam lines and the main feedwater lines. Your evaluation should demonstrate conformance with our requirements that:

a. Those compartments and tunnels which house the main steam lines, the feedwater lines, including the isolation valves for these lines, are designed to withstand the environmental effects (pressure, temperature and humidity) and the potential flooding resulting from a postulated crack equivalent to the flow area of a single-ended pipe rupture in these lines.

b. The essential equipment located within these compartments, including the main steam line isolation valves and the feedwater valves and their associated valve operators, are capable of operating in the environment resulting from the crack postulated in Item (a) above.

c. If the forces resulting from this postulated crack could cause the structural failure of these compartments, the consequent failure of these compartments will not jeopardize the safe shutdown of the plant.

d. The remaining portion of the pipe in the tunnel between the outboard safety valve and the Turbine Building meet the guidelines of Branch Technical Position APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment", with respect to the stress levels in this portion of the pipe and with respect to the location of the postulated break points.

We further require that you submit an analysis of the subcompartment pressure buildup following a postulated pipe break, including the structural evaluation of the affected subcompartments, to demonstrate that the design of the pipe tunnel conforms with our positions as stated above. If you cannot demonstrate conformance with our positions in this matter, indicate any design changes which may be required to comply with our positions. This evaluation should demonstrate that the methods used to calculate the pressure transient in the subcompartments outside of the primary containment are

the same as those used for subcompartments inside the containment for postulated pipe break. Demonstrate that the margin against a structural failure resulting from the pressure transient, are the same as those in subcompartments inside the primary containment. If you propose to use methods of analysis for subcompartments outside of containment which are different from those used inside containment, demonstrate that the methods of analysis for subcompartments outside containment assure adequate design margins. Identify the computer codes and the assumptions regarding the mass and energy release rates which you used in your analysis.. Provide sufficient design data so that we may perform independent calculations.

Response:

The compartments and tunnels which house the main steam lines and the reactor feedwater lines, including the isolation valves for these lines between the primary containment vessel and the turbine generator building, are the main steam tunnel in the reactor building and the main steam tunnel extension in the turbine generator building. Overpressurization of the main steam tunnel and tunnel extension due to a postulated pipe break is prevented by venting the main steam tunnel and tunnel extension to the turbine building, by way of the tunnel extension, and to the atmosphere, by way of the ventway structure. The following sections, tables and figures address, either totally or in part, the main steam tunnel, ventway and tunnel extension:

a. 3.6.1.18.3.1, 3.6.1.18.3.2, 3.6.1.20, 3.8.4.1.1.4, 3.8.4.1.3, 3.8.4.3.3f, 3.8.4.4.1

b. Tables 3.6-11 through 3.6-17

c. Figures 1.2-5, 1.2-6, 3.6-6g through 3.6-6k, 3.6-38, 3.6-39, 3.6-40a, 3.6-40b, 3.6-44, 3.6-49, 3.6-123 through 3.6-146, 3.8-2, 3.8-30 through 3.8-33, 3.8-38, 3.8-39, 3.8-54, 3.8-55.

An evaluation of the environmental effects resulting from a postulated pipe break in the main steam line or the reactor feedwater line demonstrates conformance with NRC requirements, as described in the following paragraphs a, b, c and d, which paragraphs correspond to paragraphs of the same designation in the question:

a. The main steam tunnel, including the blast door, removable concrete plugs, ventway and tunnel

extension, are designed to withstand the environmental effects of the predicted pressure, temperature and humidity and the potential flooding effects (described in paragraph b below) resulting from a postulated crack equivalent to the flow area of a single-ended pipe break in the main steam line or the reactor feedwater line.

b. The active essential equipment, including the main steam isolation valves and the feedwater valves and their associated valve motor operators, are capable of operating in the environment resulting from the crack postulated in (a) above. Equipment listed in Tables 01.011-1 and 010.011-2 have been appropriately included in the Environmental Qualification Report referenced in Section 3.11 with qualification information to the listed criteria.

Table 010.011-1 and Table 010.011-2 list the essential equipment in the main steam tunnel and tunnel extension. There is no essential equipment in the ventway. The tables compare the maximum qualified environmental conditions to the maximum predicted environmental conditions (pressure, temperature, humidity) resulting from the crack postulated in (a) above.

The main steam tunnel and tunnel extension face no potential flooding problem resulting from the postulated crack in the main steam line, because the closure of the main steam isolation valve terminates flow from the reactor side of the main steam line within a maximum closure time of 5.5 seconds, as discussed in the response to Question 010.013-6.

With regard to flooding resulting from a postulated reactor feedwater line crack occuring in the tunnel extensions water will flow directly upon the mezzanine floor at elevation 471'-0" in the turbine building by way of the opening in the tunnel extension at elevation 501'-0". However, the water is eventually removed by the turbine building drainage system. If the postulated reactor feedwater line crack occurs in the main steam tunnel rather than in the tunnel extension, the water will flow through the openings created by the displacement of the blow-out panels at the north end of the main

steam tunnel and at the north end of the main steam tunnel east wall. The flow-out panels are designed to blow off at a differential pressure of 0.5 psi, as noted in 3.6.1.20.3.2. If the blow-out panels were to fail to open, in the case of a smaller break, the blow-out panels will open under the water pressure developed by the water accumulated on the tunnel floor, before the flood level reaches any safety-related equipment. The water will accumulate to a height of 15-inches before failure of the blow-out panels and consequent release of the headwater occurs.

c. The forces resulting from he postulated crack do not cause structural failure of the tunnel, tunnel extension or the ventway.

d. The portion of the pipe in the main steam tunnel building meets the guidelines of Branch Technical Position APCSB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment", with respect to the stress levels in this portion of the pipe and with respect to the location of the postulated pipe break points.

The method used to calcuate belowdown is discussed in 3.6.1.20.1.3. The mechanism which terminates the blowdown is discussed in the response to Question 010.013.

An analysis of the subcompartment pressure buildup in the main steam tunnel, ventway and tunnel extension following a postulated main steam line crack (equivalent to the flow area of a single-ended pipe rupture) in the main steam tunnel or the tunnel extension, and verification of the structural adequacy of the tunnel, ventway and tunnel extension are discussed in 3.6.1.20.3. The subcompartment analysis following the postulated crack in the reactor feedwater line (equivalent to the flow area of a single-ended pipe rupture) is not discussed because, in comparing the postulated crack analyses of both the reactor feedwater and the main steam lines, the postulated main steam line crack is the limiting case as stated in 3.6.1.20.3.

The methods used to calculate the pressure transient in the main steam tunnel, tunnel extension and ventway are the same as those used for subcompartment pressure analyses inside the primary containment vessel for a postulated pipe break. The structural

design of the tunnel, tunnel extension and ventway has the same margin against structural failure resulting from the pressure transient as the structural design of the subcompartments inside the primary containment vessel.

The computer codes used in the analysis are identified in 3.6.1.20 and in 3.12. The assumptions regarding the mass and energy release rates used in the analysis are identified in 3.6.1.20.

TABLE 010.011-1

COMPARISON OF DESIGN AND PREDICTED ENVIRONMENTAL CONDITIONS
FOR ESSENTIAL EQUIPMENT IN MAIN STEAM TUNNEL(1)

ESSENTIAL EQUIPMENT IN MAIN STEAM TUNNEL (4)			EQUIPMENT ENVIRONMENTAL CONDITIONS – IN MAIN STEAM TUNNEL: MAXIMUM QUALIFIED (7)			EQUIPMENT ENVIRONMENTAL CONDITIONS – IN MAIN STEAM TUNNEL: MAXIMUM PREDICTED, FOLLOWING POSTULATED PIPE CRACK IN MAIN STEAM TUNNEL (2)			B&R REFERENCE DRAWING	REMARKS
NAME	DESIG-NATION	LOCATION IN TUNNEL	TEMP. (°F)	PRESSURE (PSIG)	HUMIDITY (%)	TEMP. (°F)	PRESSURE (PSI)	HUMIDITY (%)		
Main Steam Body Drain Shutoff Valves	MS-V-67A	South	340	45	100	307	12	100	M695,M697	
	MS-V-67B	South	340	45	100	307	12	100	M695,M697	
	MS-V-67C	South	340	45	100	307	12	100	M695,M697	
	MS-V-67D	South	340	45	100	307	12	100	M695,M697	
Main Steam Drain Block	MS-V-19	South	340	45	100	307	12	100	M695,M697	
Main Steam Isolation Valves-Outboard	MS-V-28A	South	340	45	100	307	12	100	M695,M697	
	MS-V-28B	South	340	45	100	307	12	100	M695,M697	
	MS-V-28C	South	340	45	100	307	12	100	M695,M697	
	MS-V-28D	South	340	45	100	307	12	100	M695,M697	
Main Steam Leakage Control Valves	MSCL-V-2A	South	340	45	100	307	12	100	M698	
	MSCL-V-2B	South	340	45	100	307	12	100	M698	
	MSCL-V-2C	South	340	45	100	307	12	100	M698	
	MSCL-V-2D	South	340	45	100	307	12	100	M698	
	MSCL-V-3A	South	340	45	100	307	12	100	M698	
	MSCL-V-3B	South	340	45	100	307	12	100	M698	
	MSCL-V-3C	South	340	45	100	307	12	100	M698	
	MSCL-V-3D	South	340	45	100	307	12	100	M698	
Main Steam Leakage Control Valves	MSCL-V-4	North	340	45	100	305	12	100	M698	
	MSCL-V-5	North	340	45	100	305	12	100	M698	
	MSCL-V-9	North	340	45	100	305	12	100	M698	
	MSCL-V-10	North	340	45	100	305	12	100	M698	

TABLE 010.011-1 (Continued)

ESSENTIAL EQUIPMENT IN MAIN STEAM TUNNEL (4)			EQUIPMENT ENVIRONMENTAL CONDITIONS - IN MAIN STEAM TUNNEL							
			MAXIMUM QUALIFIED (7)			MAXIMUM PREDICTED, FOLLOWING POSTULATED PIPE CRACK IN MAIN STEAM TUNNEL (2)			B&R REFERENCE DRAWING	REMARKS
NAME	DESIG-NATION	LOCATION IN TUNNEL	TEMP. (°F)	PRESSURE (PSIG)	HUMIDITY (%)	TEMP. (°F)	PRESSURE (PSI)	HUMIDITY (%)		
Reactor Feed Water Valves	RFW-V-32A(3)	South	340	45	100	307	12	100	M713	
	RFW-V-32B(3)	South	340	45	100	307	12	100	M713	
	RFW-V-65A	South	340	45	100	307	12	100	M713	
	RFW-V-65B	South	340	45	100	307	12	100	M713	
General Electric Temperature Elements	E31-NO 29A	North	340	115	100	305	12	100	E697	
	E31-NO 29B	North	340	115	100	305	12	100	E697	
	E31-NO 29C	North	340	115	100	305	12	100	E697	
	E31-NO 29D	North	340	115	100	305	12	100	E697	
General Electric Temperature Elements	E31-NO 30A	South	340	115	100	307	12	100	E697	
	E31-NO 30B	South	340	115	100	307	12	100	E697	
	E31-NO 30C	South	340	115	100	307	12	100	E697	
	E31-NO 30D	South	340	115	100	307	12	100	E697	
General Electric Temperature Elements	E31-NO 31A	South	340	115	100	307	12	100	E697	
	E31-NO 31B	South	340	115	100	307	12	100	E697	
	E31-NO 31C	South	340	115	100	307	12	100	E697	
	E31-NO 31D	South	340	115	100	307	12	100	E697	
Cables, Elec-	None	Various	340		See	307	12	100	E683,E697	0 to 3 hrs.(6)
trical, and	None	Various			Footnote	307	12	100	E683,E697	4 to 9 hrs.(6)
Instrumentation	None	Various			5	307	12	100	E683,E697	10 to 33 hrs.(6)
and Control(5)	None	Various				307	12	100	E683,E697	Remainder(6)

TABLE 010.011-1 (Continued)

NOTES FOR TABLE

(1) The main team tunnel is in the reactor building. The tunnel extension, Table Q. 010.011-2, is in the turbine generator building.

(2) The maximum predicted conditions occur during the initial few seconds.

(3) Valves RFW-V-32A and 32B are swing check valves. Their operability is not impaired by the maximum predicted pressure.

(4) There is not tubing in the main steam tunnel and tunnel extension for air lines operating instrumentation and control equipment and components or for any other applications.

(5) Electrical power cables and instrumentation and control cables were given qualification tests for nuclear power services, in accordance with IEEE Standard No. 323-1974. The tests included steam environment exposure under simulated normal operating conditions and LOCA conditions. Associated test documents are:

a. For power cables: The Okonite Company, Ramsey, New Jersey, Engineering Report No. 266, dated July 17, 1975 submitted to Burns and Roe as Transmittal 62A-00-0004, in accordance with Contract Specification 2800-62A.

b. For power cables and instrumentation and control cables: The Raychem Corp., Menlo Park, California, Report No. RABR-62B-75-028, dated November 12, 1975, submitted to Burns and Roe as Transmittal 62B-00-0094, in accordance with Contract Specification 2800-62B.

(6) LOCA ratings

(7) Qualification established in Equipment Qualification Report referenced in FSAR Section 3.11.

TABLE 010.011-2

COMPARISON OF DESIGN AND PREDICTED ENVIRONMENTAL CONDITIONS
FOR ESSENTIAL EQUIPMENT IN MAIN STEAM TUNNEL(1)

ESSENTIAL EQUIPMENT IN MAIN STEAM TUNNEL (3)			EQUIPMENT ENVIRONMENTAL CONDITIONS - IN MAIN STEAM TUNNEL EXTENSION							
			MAXIMUM QUALIFIED (4)			MAXIMUM PREDICTED,(2) FOLLOWING POSTULATED PIPE CRACK IN MAIN STEAM TUNNEL EXTENSION				
NAME	DESIGNATION	LOCATION IN TUNNEL EXTENSION	TEMP. (°F)	PRESSURE (PSI)	HUMIDITY (%)	TEMP. (°F)	PRESSURE (PSI)	HUMIDITY (%)	B&R REFERENCE DRAWING	REMARKS
General Electric Radiation Detectors	D17-N003A	North	410	279	98	313	8	98	E607	
	D17-N003B	North	410	279	98	313	8	98	E607	
	D17-N003C	North	410	279	98	313	8	98	E607	
	D17-N003D	North	410	279	98	313	8	98	E607	
Cables, Electrical and Instrumentation and Control	Data same as in Table Q. 010.011-1 and associated footnotes 4 and 5								E590,607	

(1) The main steam tunnel extension is in the turbine generator building. The main steam tunnel, Table Q. 010.001-1, is in the reactor building.

(2) The maximum predicted conditions correspond to a time duration of 0 to 2 hours. From 2 hours to 6 hours, the predicted temperature is 212°F and the predicted pressure approaches atmospheric.

(3) There is no tubing in the main steam tunnel extension for air lines operating instrumentation and control equipment and components or for any other application.

(4) Included in Equipment Qualification Report referenced in Section 3.11.

Q. 010.012

Provide the results of your evaluation of the jet impingement forces and the environmental effects, including pressure, temperature, humidity,and flooding, resulting from a postulated failure of the main steam and main feedwater systems in the Turbine Building. This evalution should address only those safety-related components, systems and structures, if any, in (or immediately adjacent to) the Turbine Building (e.g., the walls fo the Auxiliary Building).

Response:

It has been determined that the only items with safety-related functions in the Turbine Building are some RPS sensor inputs from the Main Steam System, MSIV isolation logic inputs from the Main Steam System, and the Tower Make-up Transformers located in the basement of the Turbine Building which are required to function only for the Design Basis Tornado event. This last item is remote from the steam and feedwater lines (being located at the basement grade level of the building) and has been evaluated to have adequate protection from tornado missiles and internal flooding (see the responses to question 010.025 and 010.034). In addition, there is cabling through the Turbine Building for the condensate storage tank level sensors which provide for auto-switching of HPCS from the storage tank to the suppression pool. The routing of this cabling was reviewed and found to be possibly vulnerable to pipe break effects. As a result, a redundant system for condensate storage tank level sensing, with HPCS auto-switching features to the suppression pool, was designed and installed in the Reactor Building in accordance with Seismic Category I and Safety Class 1 requirements. The new system consists of a standpipe connected to the HPCS pump suction line from the condensate storage tanks. Level sensing instrumentation monitors the condensate storage tank level and provides the auto-switching signal when required. The only other items of concern are the RPS and MSIV isolation logic sensor inputs. Due to their nature they cannot be made immune from pipe-break effects. However, no analysis has been performed of the specific effects of a steam line or feedwater break in the Turbine Building on this equipment since it has been determined that the complete loss of all this equipment could occur for these events without the loss of capability to bring the plant to a cold shutdown or mitigate the radiological consequences of such an incident even assuming a single failure in the safety systems that remain unaffected.

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Q. 010.12

Provide the results of your evaluation of the jet impingement forces and the environmental effects, including pressure, temperature, humidity, and flooding, resulting from a postulated failure of the main steam and main feedwater systems in the Turbine Building. This evaluation should address only those safety-related components, systems and structures, if any, in (or immediately adjacent to) the Turbine Building (e.g., the walls of the Auxiliary Building).

Response:

It has been determined that the only items with safety-related functions in the Turbine Building are some RPS sensor inputs from the Main Steam System, MSIV isolation logic inputs from the Main Steam System, and the Tower Make-up Transformers located in the basement of the Turbine Building which are required to function only for the Design Basis Tornado event. This last item is remote from the steam and feedwater lines (being located at the basement grade level of the building) and has been evaluated to have adequate protection from tornado missiles and internal flooding (see the responses to question 10.25 and 10.34). In addition, there is cabling for the condensate storage tank level sensors which provide for auto-switching of HPCS from the storage tank to the suppression pool. The routing of this cabling is currently through the Turbine Building, but is under design review to insure its adequate protection from accidents. Appropriate design changes will be made as a consequence of this evaluation. Accordingly, the only items of concern are the RPS and MSIV isolation logic sensor inputs. Due to their nature they cannot be made immune from pipe-break effects. However, no analysis has been performed of the specific effects of a steam line or feedwater break in the Turbine Building on this equipment since it has been determined that the complete loss of all this equipment could occur for these events without the loss of capability to bring the plant to a cold shutdown or mitigate the radiological consequences of such an incident even assuming a single failure in the safety systems that remain unaffected.

The electrical cable connected with this safety related equipment in the corridors separating the Turbine Building, Reactor Building, and Radwaste Building would be exposed to temperatures and pressure effects of a postulated failure of the main steam or feedwater lines in the Turbine Building, but the exposure conditions would be for less than the design environment requirements contained in the purchase specifications for the cable.

The electrical cable connected with this safety-related equipment in the corridors separating the Turbine Building, Reactor Building, and Radwaste Building would be exposed to temperatures and pressure effects of a postulated failure of the main steam or feedwater lines in the Turbine Building, but the exposure conditions would be for less than the design environment requirements contained in the purchase specifications for the cable.

No other safety-related equipment is located in an area which would be vulnerable to the environmental effects of a pipe break in the Turbine Building. The only safety-related structures adjacent to the Turbine Building are the Reactor Building and Radwaste-Control Building. A pipe break in a main steam or feedwater line in the Turbine Building would result in transitory pressurization of the corridors between the Turbine Building, Reactor Building, Radwaste-Control Building and Diesel-Genrator Building. Air and steam would be forced into these corridors through openings in the south wall of the Turbine-Generator Building, and through the seismic gap between the Turbine Building, Reactor Building and Radwaste-Control Building. No compartmental pressurization analysis is required to determine peak pressures and temperatures in the corridors due to the large volume of the Turbine Building, and the fact that the metal siding and exterior doors into the Turbine Building are not leak-tight and are not designed to withstand more than a minimal pressure differential, the peak pressures seen by the reinforced concrete walls of the Reactor Building and Radwaste-Control Building would not exceed the structural capacity of the walls. The doors to the control room are low-range blast doors, designed to withstand a pressure differential of 3 pounds per square inch, which is considered adequate to maintain control room habitability as discussed in 3.6.1.12.

It should be noted that the response to this question is directed towards the Turbine Building as a whole and does not cover the steam tunnel. The response to question 010.011 will address this area.

Q. 010.013

For the postulated pipe breaks, you have not provided the information required to determine:

1) The mechanism which terminates the resulting blowdown; or,

2) The period of time over which blowdown occurs.

Accordingly, for each postulated pipe break or leakage crack, indicate the time over which blowdown occurs and identify the mechanism which either terminates the blowdown or limits the amount of blowdown flow. These mass and energy flow rates will be used to evaluate the peak pressures and temperatures in compartments and structures following a postulated break of the high energy pipes inside these structures.

Responses:

Table 7.6-7 lists leak detection system instrumentation specifications. Table 6.2-16 lists valve closure times for automatic isolation functions tied into the leak detection system. Credit is taken for automatic isolation if the system capability is not affected by the postulated pipe break, or assumed as a single active component failure. Check valves close on reversal of flow in a fraction of a second. In all cases, the blowdown terminates when the inventory of fluid in the line is exhausted.

Where blowdown flow is not automatically terminated by isolation valves or check valves as described above, the duration of the blowdown event as the inventory of fluid in a line is exhausted is not considered in the analysis of peak compartmental pressure and temperature. To evaluate the peak pressures and temperatures in compartments and structures following a postulated break of the high energy pipes inside the structures, the blowdown analysis is extended far beyond the initial transient until the blowdown flow becomes steady or decreases continuously. The duration of the analysis is therefore sufficient to correctly predict the peak pressures and temperatures in these compartments and structures.

For a postulated pipe break or leakage crack in the main steam lines outside the primary containment, the flow from the reactor side of the break is terminated by the closing of the main steam isolation valves located in each of the four main steam lines. The main steam isolation valves start to close at 0.5 seconds after the break and are fully closed at or prior to 5.5 seconds after the break, as given in Table 15.6-6.

For a postulated break or leakage crack in the reactor feedwater lines outside primary containment, the flow from the reactor side of the break is terminated by closing of the check valves in each of the two reactor feedwater lines. The check valves start to close when the direction of flow reverses, and the flow from the reactor side of the break is therefore terminated within a fraction of a second.

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Q 10.14
(3.6)

You state in 3.6.1.1.1 of the FSAR, that fluid piping systems which the staff would classify as high-energy lines are considered by you to be moderate-energy systems if: (1) their fluid temperatures are below 200°F and; (2) the fluid pressure is generated by centrifugal pump instead of a fluid reservoir. (The staff classification system states that the fluid temperature must be less than 200°F and the fluid pressure must be less than 275 psi for a system to be designated as moderate-energy). Accordingly, demonstrate that these systems do not contain enough energy to cause pipe whip. Additionally, provide justification for your analysis of flooding based on the moderate-energy crack criteria rather than basing your analysis on the full break required by the high-energy break criteria.

Response:

The energy of the blowdown fluid from a break in a pressurized fluid system is a function of the pressure at the exit plane, the mass flow rate and the area of the fluid jet. The blowdown process of the 200°F water from high pressure to atmospheric pressure can be considered as adiabatic. Since the water is subcooled, it will not flash during the compression transient of the blowdown. Therefore, the water jet remains in the liquid phase and behaves like an incompressible fluid.

At the beginning of the decompression transient, immediately following the break, a decompression wave is formed and travels through the fluid at sonic velocity (approximately 5,142 ft/sec) to the pressure source which, in this case, is the centrifugal pump. Due to the reduction in required head, the flow rate accelerates rapidly increasing accordingly to the characteristics of the piping system until a new equilibrium is established. Since the system operating pressure is derived solely from the centrifugal pump, the complete system is depressurized after the break and the energy supplied to the pump is completely transmitted to the fluid in terms of velocity head. For an incompressible fluid in an open system, the energy of the water jet is proportional to the velocity head only. Hence, the thrust of the water jet from a break in this class of piping systems may be calculated at the exit plane of the jet using the following formula:

$$F \doteq \rho A V^2 / g_c$$

where:

F = jet force normal to target, Lb_f

ρ = fluid density, Lb_m/ft^3

A = flow path cross-sectional area, ft^2

V = velocity of jet, ft/sec

g_c = 32.179 Lb_m - ft/Lb_f - Sec^2

Maximum jet thrust may be obtained at the exit plane of the pump. Assuming a friction coefficient of 1.5 to include only contraction and expansion losses, two representative examples representing the highest head and highest flow cases in the plant are presented below to demonstrate that this class of high pressure systems do not cause pipe whip events if the system pressure is derived from a centrifugal pump.

	CRD Pump Discharge	Condensate Pump Discharge*
Piping Designation	2" CRD (2) - 4	20" Cond (2) - 1
Pipe Schedule	160	40
Operating Pressure	1,439 psig	142 psig
Operating Temperature	100^{o}F	109.4^{o}F
Jet Flow	284 gpm	19,000 gpm
Jet Force	50 Lb_f	1,800 Lb_f
Jet Pressure at Break Plane	22.3 psi	6.5 psi

* By pressure and temperature criteria, this is classified as a moderate energy break. However, in Question 110.18 the NRC requested consideration of condensate piping as a high energy system.

For the maximum 10' span which exists between the pipe supports, the above pressures result in pipe stresses which are below the minimum for formation of a plastic hinge and thus, pipe whip will not occur.

Therefore, for piping systems with system pressure derived from a centrifugal pump, the system is treated as a moderate energy system and flooding analysis is performed based on postulated flow from a controlled leakage crack.

Q. 010.015
(RSP)

We require that you modify the main steam line isolation valve leakage control system (MSIV-LCS) to satisfy the staff positions contained in Regulatory Guide 1.96, Rev. 1, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants", June 1976. Specifically, we require that:

a) The design of the MSIV-LCS permits its actuation within 20 minutes after a postulated loss-of-coolant accident.

b) The leakage control system for the valve stems on the main steam line be designed to the same standard as the MSIV-LCS, and

c) Operation of the MSIV-LCS during normal plant operation be prevented by inter-locks capable of functioning after a postulated single failure in the inter-locking system.

Response:

a) See revised page 6.7-2.

b) See revised page 6.7-11.

Also: (direct response)

Stem packing leakage from the outboard main steam isolation valves is directed to equipment drain funnels located in the steam tunnel. The leak-off piping is classified as Safety Class 2 up to and including the first manual block valve. As stated in 6.7.3m, leakage from the packing seals large enough to pressurize the steam tunnel and blow out the water seal traps in the equipment drain system would vent into areas of the reactor building for subsequent processing by the standby gas treatment system. Refer to Figure 3.2-2, Zone G2, which depicts the stem packing leakage piping.

c) Refer to Question 031.076 response.

Q. 010.016
(9.0)

Identify all safety-related equipment that could be exposed to, or affected by, dust storms. Describe how you propose to assure the proper functioning of this equipment during dust storms. Provide a description of the methods which will be used to prevent the blockage of vital air supplies to safety-related equipment (e.g., clogging of the air filter of the Diesel Generators). In your response to this question, provide a cross-reference to your response to 372.008.

Response:

Essentially all safety-related equipment that could be affected by severe dust storms are contained within plant areas served by the HVAC systems for the reactor building, control room, cable spreading room, critical switchgear areas, standby service water pumphouses and diesel generator building. The only safety-related system exposed directly to severe dust conditions are the service water spray ponds.

a) The normal air intakes for the reactor building and control room, cable spreading room, and critical switchgear are located 130 feet and 85 feet above ground level, respectively. At these intake locations the dust loadings will be 10 to 15 percent of ground level dust loads (See Figure 2.3-5 and the response to Question 372.008 for representative dust loads). All intake air is processed through either automatic roll type filters or replaceable filter elements in the air handling units before entering the air distribution systems for the reactor building, control room, cable spreading room, and critical switchgear areas. An air washer is also included in the reactor building air handling unit. Pressure differential across the filter units is annunciated when filter replacement is required.

   With the intake locations and filtration, discussed above, the amount of dust entering the reactor building, control room, cable spreading room and critical switchgear areas will not degrade the operating capability of safety-related equipment in these areas.

b) The standby service water pumphouses have unfiltered outside air intakes and some dust may be expected to enter the pumphouses during severe dust conditions. The amount of dust, however, should be limited since the pumphouse HVAC systems will be shut down during normal plant operations with the intakes and exhaust openings restricted by dampers.

The dust loading of the air drawn into the pumphouses, when the HVAC systems are operating, should be less than maximum gound level dust loads since the air intakes are located above the service water spray ponds and feed into a plenum before entering the intake fan. Any dust which enters will settle out within the pumphouses without blocking vital air passages.

Entry of dust into the service water pumphouse will not affect the operation of safety related equipment. Any equipment that could be affected by dust is either provided as sealed units, located in dust proof cabinets or protected by dust proof coatings.

c) The diesel generator building outside air intake is located at grade level with air filters located within the building at 15 feet above grade.

The common air filter bank processes all ventilation air into the diesel generator building. During a worst case dust storm, as defined in 2.3.1.2.1.5.2, the maximum estimated dust load will be 8.9 mg/m^3 for an 18 hour duration. After particle impaction and re-entrainment (due to intake louvers) is accounted for, the calculated dust load to the filters is 6.44 mg/m^3. Without taking any credit for particle settling in reduced velocity area before filter bank the filter will be subjected to a maximum of 0.231 $\#/F^2$ of dust.

The filter bank consists of two (2) filters (prefilter and final filter) in series with a common pressure switch which will alarm when filters require changing. The prefilter normal maximum resistance is 0.50" W.G. (equal to 0.047 $\#/F^2$).

The final filter normal maximum resistance is 1.00" W.G. (equal to 0.142 $\#/F^2$). During severe dust storm conditions the prefilter can be loaded to 0.129 $\#/F^2$ or 1.00" W.G.

During the postulated dust storm an initial filter alarm would require a complete filter change followed by a maximum of two (2) prefilter changes as filter alarms are annunciated.

The 18 hours loading on the diesel air filters is calculated to be 2102.4 grams. The capacity of the air filters is 5000 grams or 2.378 times the severe dust storm loading.

This response is an elaboration of the response given in part (d) to Question 40.26.

d) The ultimate heat sink transient analysis was performed assuming 6" of sedimentation at the bottom of the spray ponds (see 9.2.5). In addition, no credit is taken in the analysis for the volume of water within the sand traps which prevent sedimentation from being swept into the pump pits.

Q. 010.017
(9.1.2)

The design of your spent fuel rack includes a neutron absorbing material encapsulated in stainless steel. However, recent experience at some spent fuel pools has shown that the stainless steel cladding may bow out due to the internal pressure of gases generated by the irradiation of the neutron absorbing material in the spent fuel pool. This bowing of the steel cladding has caused the spent fuel assemblies to become lodged in the spent fuel racks. Accordingly, describe the method (e.g., venting the stainless steel plates to release any evolved gases) you propose to prevent this from occurring in the WNP-2 spent fuel pool.

Response:

Bowing of the steel cladding is not expected to occur since the neutron absorber plates utilized in the WNP-2 racks have been shown through testing not to offgas when irradiated by a gamma source. These plates manufactured by Electroschmelzwerk-Kempten (ESK) differ substantially in composition and manufacturing process from the type plates which underwent decomposition at Connecticut Yankee. Because of the nonoffgassing characteristic of these plates, venting of the racks is not planned. For additional information on the offgassing tests, refer to the response to Question 010.018 (9.1.2).

Q. 010.018
(9.1.2)

In 9.1.2 of the FSAR, you list the test results involving radiation, thermal, seismic and borated water testing of the boron carbide plates. Describe the procedures used for these tests. Alternatively, provide a cross-reference to any of these test procedures which have previously been accepted by the NRC staff on another application.

Response:

When the FSAR was originally written, the manufacturer of the boron carbide plates had not been identified. Accordingly, data from previously licensed plates was used based on a program description and results of the qualification tests conducted on boron carbide neutron absorber plates submitted to the NRC under the Connecticut Yankee docket 50-213, letter, D.C. Switzer to R.A. Purple, dated April 15, 1976. Subsequently, ESK was selected as the manufacturer of the plates. With the exception of the full scale seismic test, essentially all described tests have been performed by ESK for the plates of their manufacture. Because of the similarity in physical characteristics with the plates previously tested and because Modulus of Rupture tests show plates will withstand two times calculated seismic stresses, repetition of shaker table testing was not deemed necessary. Test results for the ESK plates were submitted to the NRC under the Kewaunee docket 50-305 in a letter, E.W. James to V. Stello, dated September 5, 1978.

As a result of our decision to use ESK plates, 9.1.2 has been revised.

Q. 010.19
RSP
(9.1.2)

In 9.1.2.3.3 of the FSAR, you state that the interlocks which prevent the 125 ton crane in the reactor building from traversing the spent fuel pool, are occasionally by-passed. This by-passing is unacceptable. Accordingly, we require you to modify your procedures so that the interlocks on the reactor building crane prevent the crane from traversing over the spent fuel pool whenever there is spent fuel in the pool.

Response:

Regulatory Guide 1.13, c.3 allows for movement of loads necessary for fuel handling over the spent fuel. Occasionally it is necessary to operate the reactor building crane over the spent fuel pool in conjunction with maintenance of fuel storage and fuel handling facilities, or other activities associated with fuel handling and storage. Therefore it is necessary to retain the ability to bypass the interlocks and use administrative control procedures under those conditions. Movement of objects in excess of the rack design drop load (one fuel assembly at four feet above the top of the fuel rack) will be prohibited. The electrical interlocks are bypassed only by actuation of a cab-mounted key-lock switch.

See revised 9.1.2.3.3 Appendix c.3 page 11, and revised FSAR Figure 9.1-17 which shows the interlock-controlled restricted area for crane travel over the spent fuel pool.

Q. 010.20
RSP
(9.1.2)

In 9.1.2 of the FSAR, you state that a portion of the fuel handling building above the refueling floor is constructed of sheet metal. Accordingly, we require you to demonstrate that the spent fuel pool is housed in a Seismic Category I structure which can withstand the impact of tornado missiles.

Response:

Table 3.2-1 states that the Reactor Building is designed to Seismic Category I requirements. Section 3.5.1.4 states that Seismic Category I structures are designed to include the effects of missiles generated by the design basis tornado. Section 3.8.4 provides details of the design features of the Reactor Building and spent fuel storage pool. Tables 3.8-15 and 3.8-16 provide the load combinations and load factors used in design of Seismic Category I structures. Section 3.5.1.4.1.a discusses the tornado missile-resistant design features of the Reactor Building. Section 3.3.2 discusses the design features of the Reactor Building for tornado wind loading. As stated in 3.5.1.4.1.a and 9.1.2.3.5, which reference GE Topical Report APED-5696, the design basis tornado missile for the refueling floor has been evaluated and found to not have sufficient energy to damage the spent fuel or the equipment and structures in the pool.

Q. 010.021
(9.1.3)

Provide a cooling system and a source of makeup water for the spent fuel pool which are both designed to Seismic Category I criteria in accordance with the staff positions contained in Regulatory Guide 1.13, Revision 1, "Spent Fuel Storage Facility Design Basis," December 1975.

Response:

See the response to Question 010.056.

Q 010.22
(9.2.1)

Identify which valves are used to isolate that portion of the plant service water system which is not designed to Seismic Category I criteria from that portion which is designed to these criteria. Provide a failure modes and effects analysis for the plant service water system, assuming a seismic event has occurred.

Response

The plant service water system (TSW) is not required for safe shutdown and accordingly is not designed to Seismic Catetory I requirements. A failure modes and effects analysis is not considered necessary. The portions of the TSW system piping in the Reactor Building have been designed to Seismic Category I requirements so that they will not fall and damage safety related equipment.

The standby service water system (SW) is used for safe shutdowns and is designed to Seismic Category I criteria. The SW is discussed in 9.2.5. The plant service water system (TSW) and the standby service water system (SW) are independent systems and are not connected, therefore there are no valves which are used to isolate these systems from each other.

Q. 10.23
RSP
(9.2.5)

Provide the results of your analysis of the capability of the ultimate heat sink to absorb heat over a thirty-day period following a postulated design basis accident. Indicate the total heat absorbed in the ultimate heat sink, including the sensible heat, the station auxiliary system heat, and the decay heat released by the reactor core. In particular, provide the following information in both tabular and graphical formats:

a. The total integrated decay heat.

b. The heat rejection rate and the integrated heat rejected by the station auxiliary systems, including all operating pumps, ventilation equipment, diesels and other heat sources.

c. The heat rejection rate and integrated heat rejected due to sensible heat removed from the containment and the primary system.

d. The total integrated heat rejected; i.e., the sum of the Items (a), (b) and (c).

Additionally, provide the following information:

e. The maximum allowable temperature of the inlet water taking into account the rate at which heat must be removed, the cooling water flow rate, and the capabilities of the respective heat exchangers.

f. The required and available net positive suction head (NPSH) at the suction lines of the service water pumps at the minimum water level of the ultimate heat sink.

This analysis should demonstrate the capability of the ultimate heat sink to provide: (1) an adequate water inventory; and (2) sufficient heat dissipation which will limit the essential cooling water operating temperatures within the design ranges of system components. In this regard, we require you to use the methods contained in Branch Technical Position ASB 9-2 "Residual Decay Energy for Light Water Reactors for Long Term Cooling," when evaluating the residual decay energy

release rate from the reactor core due to fission product decay and heavy element decay. Assume an initial cooling water temperature based on the most adverse conditions possible during normal operations. The meteorological conditions should be established following the guidance contained in Position C.1 of Regulatory Guide 1.27, Revision 1, "Ultimate Heat Sink for Nuclear Power Plants," March 1974.

Response:

See revised 9.2.5 which provides the revised results of the analysis including the information requested above. This revision also addresses the concerns of Question 371.6.

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Q. 010.24
RSP
(9.2.5)

We require that you protect the sprays in the ultimate heat sink from the effects of tornados and tornado missiles.

Response:

As discussed in 3.3.2.3, the WNP-2 UHS design provides for continuous water make-up to the spray ponds in the event that both the spray systems are rendered inoperable due to tornado missiles. Therefore, the sprays are not required to be protected from the effects of tornado missile since an alternate UHS operating mode (continuous Make-up) is available which is protected from the effects of tornadoes and tornado missiles.

Q. 010.25
(9.2.5)

In the event that a tornado siphons water from the ultimate heat sink (UHS), the make-up water pumps will replenish the UHS. Demonstrate that the transformers located in the turbine building and the electric cabling which are both required to operate the make-up pumps, are protected from tornados and tornado missiles.

Response:

As described in 3.3.2.3, the TMU transformers (TR-75-72 and TR-85-82) are located at gound level in the southeast corner of the turbine building where they are protected by the exterior walls of the turbine building, the reactor building to the south, the service building to the east, and other reinforced concrete interior walls to the north and to the west, and are therefore not considered vulnerable to tornado missile impact. As described in 3.5.2, electrical cabling to the TMU pumphouse is buried at sufficient depth in compacted backfill to provide protection against tornado missiles. Electrical cabling from each transformer is routed separately to two switchgear units at ground level in the southwest corner of the turbine building. For missile trajectories which would jeopardize the TMU transformers, associated cabling, and switchgear, the exterior walls of the turbine building provide adequate protection against design basis missile penetration and spalling.

Q. 010.26
(9.2.5)

In 9.2.5 of the FSAR, you state that the two ponds which comprise the ultimate heat sink are connected by a siphon that allows water to flow from one pond to the other. Demonstrate that a failure in this siphon line, or in one of the ponds, will not result in draining of both ponds.

Response:

The siphon between the two ponds is a Seismic Category I, Quality Group C 30 inch pipe, whose centerline is 4 ft. 6 in. below the normal water level of the spray ponds.

Therefore a siphon line failure would be considered a passive failure. Applying single failure criteria indicates that if the siphon failed then both SW loops would be operating, thus keeping them at the same level. If one of the SW loops fails, then an additional failure of the passive siphon is not considered credible.

The spray ponds are Seismic Category I structures located below grade with continuous waterstops in all joints and bounded with Quality Class I high density backfill. Both ponds together form the Ultimate Heat Sink, a concept which has been accepted on other plants that only have a single pond which contains the redundant spray networks. Failure of either Pond A or Pond B will result in drainage of the other pond, which results in the same consequence if the WNP-2 UHS were a single pond design. However, as described above and in 3.8.4.1.5 the spray ponds have been conservatively designed to preclude pond failure.

Q. 010.27
RSP
(9.2.7)

We require that you protect the standby service water system from tornado missiles.

Response:

The standby service water system (except for the spray pond spray piping) is protected from tornado missiles. The structures which house the standby service water systems (Reactor Building, DG Building, Control Building, and SW Pumphouse) have been designed to withstand design basis tornado generated missiles as described in 3.5.1.4.1.

Buried portions of the standby service water system are protected from tornado missiles as described in 3.5.2.

See the response to question 10.24 as to why it is not necessary to protect the spray pond spray headers from tornado missiles.

Q. 010.028
(9.3.4)

Describe how flooding of safety-related equipment due to back-flooding through the equipment and floor drainage system, is prevented. Demonstrate that those portions of the drainage system necessary to prevent backflooding (e.g., check valves) are designed to Seismic Categorv I criteria and that their system function will be maintained, assuming a single active failure.

Respone:

It is assumed that the question is directed to FSAR 9.3.3.2.2.1, Reactor Building Floor Drains, and not 9.3.4, Chemical and Volume Control System.

As shown on Figure 9.3-8, the floor drain piping in the reactor building drains to one of four sumps listed below.

FLOOR DRAIN SUMP	ROOM LOCATIONS	ROOMS SERVED
FDR-R-1	RHR A Pump Room	RCIC RHR A
FDR-R-2	RHR B Pump Room	RHR B
FDR-R-3	HPCS Pump Room	HPCS CRD
FDR-R-4	RHR C Pump Room	LPCS RHR C

Each of the four downcomers is equipped with instrumentation which alarms in the control room to tell the operator at which elevation an excess of water is collecting in the downcomer. Each sump is equipped with level instrumentation which: 1) controls the sump pumps, 2) alarms in the control room (on high sump level), and 3) initiates closure of the isolation valves in the piping between interconnected rooms. Not currently shown on Figure 9.3-8 are Class 1E level instrumentation to be installed just above floor level in each ECCS pump room. This instrumentation will alarm in the control room.

The floor drain system is analyzed against the potential sources of flooding within the reactor building, i.e., pipe break outside containment and passive failures in the ECCS during post-LOCA long term cooling. Using the acceptance criteria for either event (Standard Review Plan 3.6.1 and Reactor Systems Branch Technical Position, Leak Detection Requirements for ECCS Passive Failures), the floor drain system design is acceptable in mitigating the consequences of flooding ECCS pump rooms.

The effects of pipe breaks outside containment are addressed in 3.6.1.11.4, i.e., ruptures in fluid systems have no effect on the ability to bring the reactor to a cold shutdown condition. Single random active failures are assumed in the analysis and credit is taken for systems not affected by the flooding. As stated in 3.6.1, these assumptions and the approach taken are consistent with the guidance of Branch Technical Position APCSB 3-1. This is in conformance with the criteria of Standard Review Plan 3.6.1, March 1975.

The effects of passive failures in the ECCS during post-LOCA long term cooling is addressed in the response to Question 212.003. The largest passive failure has been identified as the total failure of an RHR pump seal and it is equivalent to a 23 gpm leak. Class 1E instrumentation in each ECCS pump room will detect the leak and give the operator at least 44 hours to identify and isolate the passive failure before it has any additional adverse effects on ECCS operation.

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Q. 010.29

Demonstrate that the heating, ventilating and air conditioning systems (HVAC) for the engineered safety features are protected from tornado missiles.

Response:

Except for standby service water piping, control room remote air intake piping and control room remote air intakes, the HVAC systems serving engineered safety features are all located within reinforced concrete sturctures designed to withstand the effects of design basis tornado missiles. These structures include the reactor building, radwaste and control building, diesel generator building and standby service water pumphouses. Design of the buildings, including protection for HVAC system air intakes and exhausts, is discussed in 3.5.

The standby service water piping runs between the standby service water pumphouses and the reactor building and supplies water to the cooling coils of critical HVAC equipment during a design basis accident. The control room remote air intake lines run between the remote air intakes and the radwaste and reactor buildings, respectively, to supply control room pressurization and makeup air during accidents involving radioactive releases. Since these piping runs are all covered to a depth of over five feet with Class I compacted earth fill, they are adequately protected from tornado missiles as discussed in 3.5.2.

The two control room remote air intakes are over 200 feet from any major plant structure and are located in a northwest and southeast direction, respectively, from the turbine-generator, radwaste and reactor buildings complex. The intakes are of reinforced concrete construction designed to withstand design basis tornado missiles.

The roof slab of each intake is 12 feet square and 2 feet thick with a grated 3 foot square opening for the intake air. Eight 4" pipes surround the intake opening and serve both as barriers and as alternate air intakes in the event the grated opening is blocked or damaged. The top of the roof slab is 15 inches above the surrounding grade level. The walls and floor of the intake structure are 18 inches thick and are buried to a depth of approximately 9 feet in Class I compacted earth fill. An internal barrier is provided as additional protection for

the intake piping and to assure an unobstructed path for the flow of control room air. This barrier, which is 4 feet wide and 18 inches thick, is supported from two sidewalls and two one by two foot columns. Details of the intake structure are shown on Figure 3.5-52.

Q 010.30
(9.4.0)

You state in the FSAR that the outdoor design temperature range for the HVAC is 0°F to 105°F. However, you also indicate on Page 9.4-2 of the FSAR, that the extreme outside temperature range is minus 27°F to 115°F. Provide the results of your analysis which demonstrate that the functional capability of safety-related equipment will not be impaired by the outdoor temperatures which would occur during these extreme meteorological conditions. The effect of extreme low temperatures on safety-related equipment located outdoors should also be discussed.

Response

This question has been addressed in 9.4 (Page 9.4-2). Even though the normal outside temperature range for the design of the HVAC systems was 0°F to 105°F, as stated in the FSAR, operation at the extreme conditions of -27°F and 115°F were also evaluated. Where necessary, equipment was selected to assure operation of safety related systems during the extreme conditions.

As discussed in 9.4.7, 9.4.8 and 9.4.10, the heating equipment for the diesel generator building, diesel generator cable corridor and standby service water pumphouses are capable of maintaining temperatures at 35°F or above during the extreme -27°F conditions. In sizing heating equipment for primary operating areas of the plant, such as, the control room, reactor building or turbine-generator building, no credit was taken for heating available from plant lighting or operating equipment. Including these additional heating sources, even in a shutdown mode, the exisiting heating equipment is adequate to maintain the areas served above minimum set temperatures during the extreme cold condition.

The extent and duration of any room temperature increases which may result during operation at the extreme summer temperature of 115°F with the existing cooling systems, will not be sufficient to degrade the operating capability of any safety related equipment.

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Q 010.31
(9.4.4)

The Radwaste Building chilled water system which is not designed to Seismic Category I criteria, is connected to the HVAC system at the control room, and to the standby service water system. Provide your analysis which demonstrates that the potential failure of the Radwaste Building chilled water system during an earthquake will not cause an unacceptable degradation of the control room HVAC system and the standby service water system.

Response

The Radwaste Building chilled water system is completely isolated from the standby service water system. Both control room air handling units are Seismic Category I and are provided with two "N" stamped cooling coils. One coil is connected to the Radwaste Building chilled water system and the other coil is connected to the standby service water system. Failure of the chilled water system will not adversely effect control room or the standby service water system. Please see 9.4.1.2.1 for additional information.

Q 010.32
(9.4.7)

Demonstrate that the ventilation system of the Diesel Generator fuel oil pump room is designed to Seismic Category I criteria, and receives power from the Class 1E buses.

Response

Please see 9.4.7.3, Figure 9.4-7 and Figure 8.3-1d for requested information.

Q 010.33
(9.4.7)

Provide your analysis which demonstrates that the potential failure of the heaters in the Diesel Generator HVAC System which are not designed to Seismic Category I criteria, will not have an adverse affect on the functional capability of either the Diesel Generator or the Diesel Generator HVAC System.

Response

There are two types of heaters in the diesel generator spaces, electric unit heaters in the diesel oil pump rooms and electric heating coils in the duct systems in the diesel engine rooms themselves. The electric unit heaters in the pump rooms are Seismic Category II. These heaters are supported as Seismic Category I, however, and can fail in place without affecting any safety related equipment. Oil pump room unit heaters are used only for maintenance during cold weather for personnel comfort. The heating coils in the generator room themselves are Seismic Category I and are designated Class 1E.

Q. 010.034
(10.4.5)

Your response to Item 010.009 is unacceptable. Specifically, your analysis of flooding due to failure of the circulating water system is based on a crack whose area is equal to one-quarter of the pipe diameter times the pipe thickness (.5t X .5d). Provide an analysis of flooding due to a postulated failure of the expansion joint in the circulating water system assuming a double-ended guillotine break at this location.

Response:

See revised 10.4.5.3.

Q. 010.035
(3.4.1)

The FSAR states that the "Seismic Category I piping and electrical conduit penetrations that are below grade ... are ... not sealed against groundwater pressure." Demonstrate that the safety functions would not be compromised by water flowing into the building through these piping and conduit penetrations as the result of the following events:

a. Another compartment is flooded and water is flowing out of the building through the piping and conduit penetrations, resulting in saturated ground conditions.

b. A non-Seismic Category I tank ruptures emptying all of its contents.

Response:

Please refer to revised 3.4.1.4.2, page 3.4-4 for the information requested.

The design and installation of the boots at pipe penetrations and of the sealing material at conduit penetrations, described in 3.4.1.4.2, provide water resistant penetrations that are capable of preventing the compromise of safety functions by events such as those stated in the question.

The response to question 010.010 has been similarly revised.

Q. 010.036
(3.5.1)
RSP

It is the Staff's position that all safety-related equipment shall be appropriately protected aginast the effects of internally generated missiles in accordance with Title 10, Code of Federal Regulations Part 50, Appendix A, General Design Criteria 4. The effects of internally generated missiles such as valves stems, bonnets, control rod drive mechanisms, and high pressure accumulators impacting onto safety-related equipment must be evaluated. Appropriate protection must be provided to assure that a missile will not prevent a safe shutdown of the plant or result in uncontrolled release of radioactivity during normal operation or during the most severe design basis accidnet with the most limiting single active failure. Describe the means provided for assuring protection of safety-related equipment from all internally generated missiles.

Response:

The means for assuring protection of safety-related equipment from all internally generated missiles was submitted via letter no. G02-82-672.

Q. 010.037
(3.5.1.2)

Regulatory Guide 1.70, Revision 3, Section 3.5.1.2, requires that the structures, systems, and components protected by physical barriers shold be identified. The discussion and the figures in the FSAR do not indicate where, if it all, physical missile barriers are used.

Identify all structures, systems, and components that are protected by physical barriers. Provide a description of the types of physical barriers that are employed at your plant.

Response:

Updated 3.5 identifies all structures, systems, and components in the plant which are protected by missile barriers. Safety-related systems and equipment outside the primary containment are located in isolating compartments by division. The walls of these compartments are designed to contain all postulated missiles internal to the compartment and to protect against missiles originating outside the compartment.

External walls providing protection from missiles external to the plant are described in 3.8.4. Figure 3.5-36 illustrates these walls. Spray ponds and piping are protected by earth cover barriers. Exterior openings for HVAC systems are protected from transmitting missiles by labyrinths of missile shield walls. These openings are listed in Table 3.6-6. Barrier design procedures will be discussed in 3.5 as follows:

I. BARRIER DESIGN PROCEDURES

The design objectives emphasize missile containment and structural integrity without secondary missile generation.

a. Concrete Barriers

Concrete missile barriers are designed in accordance with the modified Petry equation (Reference 3.5-2). In all cases, except for barriers exposed to turbine missiles, a concrete thickness of twice the penetration thickness determined for an infinitely thick slab is provided to prevent perforation, spalling or scabbing. For discussion of turbine generator missiles see 3.5.1.3.

b. Steel Barriers

The Ballistic Research Laboratories Formula (Reference 3.5-1) is used to determine penetration depths of missiles into steel barriers.

The overall response of bariers subject to impact are investigated by the use of general energy equations given in "Introduction to Structural Dynamics," J. M. Bigs (Reference 3.5-9). Upon determination of penetration depth and duration of impact, an effective dynamic force is computed. The additional calculation of the natural period of the target structure and the selection of a ductility ratio facilitates the determination of the required structural resistance. In this manner, missile impact is translated to an equivalent static load in an effort to quantify bending moments and shear. The detailed method used for predicting the overall response of missile barriers, including the forcing function method of determining ductility in structural elements and the basis for the ductility ratios used in the calculations, is provided in Appendix C of the report "Protection Against Pipe Breaks Outside Containment" (Reference 3.5-13) that was presented to and approved by the NRC.

c. Earth Barriers

When the protective barrier is of earthen origin, the soil penetration studies are based on alternate techniques. Buried safety-related piping and electrical systems required for a safe shutdown are ensured adequate protection from tornado generated missiles. Analysis of potential damage is performed using the "Tornado Design Considerations for Nuclear Power Plants" by Bates and Swanson, 1976 (Reference 3.5-8). The analysis procedure neglects soil interlocking under a suddenly applied load and ignores lateral soil resistance. A five-foot embedment depth is calculated to be acceptable to ensure pipe integrity.

d. Applications

Examples of barrier design are as follows:

Steel covers for manholes containing cabling for safety-related equipment required for safe

shutdown are designed to withstand tornado generated missile impact and associated wind-pressure. These 2'-9" circular steel plates are designed using conventional elastic analysis and design methods for determining stress and strain. The design adopted uses two 1-1/8-inch plates of ASTM A 514 steel plate to prevent penetration and blowout.

The reactor building railroad airlock exterior doors and the standby service water pumphouse exterior equipment doors are designed and certified by the manufacturer to withstand the effects of tornado generated exterior missiles as described in 3.5.1.4.

All other doors in Seismic Category I and safety-related structures are not designed to withstand the effects of the missiles described in 3.5.1. These doors are backed up, wherever missile protection is required, with reinforced concrete walls forming a labyrinth behind the door. Similarly, louvers in exterior walls, which are vulnerable to missile penetration, are backed up by reinforced concrete plenums or walls.

Based upon the selection and description of missiles cited in 3.5.1, the interaction of missiles with structural elements is determined and the results are given in Table 3.5-5. The tabulations assume the missiles to impact at the most vulnerable point of a structure or component (e.g., at the center of a slab).

The reactor protection system motor generator sets flywheels located in the critical DC switchgear rooms at elevation 467'-0" in the radwaste building were analyzed and determined to be credible missile sources, with the potential consequences affecting the safe shutdown of the plant. Barriers were constructed around these flywheels of steel and aluminum honeycomb material, which were designed to contain the credible missiles.

Q. 010.038
(3.5.1.2)

Section 3.5.1.1.2 of the FSAR states that missile trajectories are selected to encompass the most adverse conditions. It is not clear from the information provided in the FSAR what the trajectories of the credible primary missiles would be and what systems might be disabled by the missiles.

Provide the bases for selection of the probable missile trajectories and show the trajectories on the appropriate FSAR figure. Include a discussion on the system, component, or structure that could be damaged or disabled by a missile. The extent of damage from each missile should be discussed.

Response:

The basis of selection of the probable missile trajectories utilizes SRP 3.6.1 and 3.6.2 which outlines the procedures for trajectories of jets. For rotating missiles, a 10° divergence angle is assumed in addition to their possible path.

A discussion of the analytical approach is provided in the response to Question 010.036 and a tabulation of credible missiles provided in updated 3.5.

Q. 010.039
(3.5.1.2)

Section 3.5.1.1.3.2 states that thermowells and sample probes do not present potential hazards as postulated missiles affecting safe shutdown.

Provide justification to support this position on the thermowells and sample probes.

Response:

Temperature or other detectors installed on piping or in wells are evaluated as potential missiles if a single circumferential weld would cause their ejection. This is highly improbable, since a complete and sudden failure of a circumferential weld is needed for a detector to become a missile. These circumferential welds were analyzed and found to have design stress intensities at least 18 times the stress intensities that will be experienced in service. Because of their highly conservative design, thermowells and sample probes are not considered credible missiles.

Q. 010.040
(3.5.1)

The FSAR states that the water lines are "...tornado-hardened." State your criteria for protecting pipes located outside buildings from tornado missiles, including depth below grade requirements and provide drawings which show all pertinent tornado protection features as necessary.

Response:

As stated in 3.5.3, buried safety-related piping required for safe shutdown is ensured adequate protection from tornado-generated missiles. Analysis of potential damage is performed using "Tornado Design Considerations for Nuclear Power Plants" by Bates and Swanson, 1967 (Reference 3.5-8). A 5-foot embedment depth is calculated to be acceptable to ensure pipe integrity.

The standby service water piping exits the pumphouses at a centerline elevation of 435'-3" and immediately turns down at a 45° angle to elevation 432', where the piping is routed to the reactor building in high relative density Quality Class 1 backfill. Grade level is at 440'-6", providing an embedment depth of over 7 feet from the top of the pipe. Where the pipe exits the pumphouses, a 1-1/2" asphaltic concrete road with a 6" base coarse and 2" leveling coarse bed provides additional protection from tornado-generated missiles. Additionally, the two standby service water loops are separated by at least 20 feet to preclude loss of redundancy. The standby service water pumphouses, shown in Figures 3.5-48 and 3.5-49, are protected from tornado-generated missiles. The standby service water piping and the tower makeup water system from the river are the only safety-related water piping systems outside of tornado protected buildings. The tower makeup system is only required in the event that the spray ring headers in the ultimate heat sink are lost in the tornado. The tower makeup piping to the river also satisfies the five-foot embedment criteria. Protection from tornadoes and tornado missiles in regard to such piping has also been previously addressed in response to Questions 010.024 and 010.027.

Though not technically a piping system in line with this question, the control room remote air intakes are, of course, located remote to tornado-hardened buildings. The intake structures themselves are tornado hardened, however, (see Figure 3.8-59) and the piping from the structures meets the five-foot embedment criteria.

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Q. 010.041
(4.6)

Demonstrate that the scram discharge system meets the criteria enumerated in the Generic Safety Evaluation Report BWR Scram Discharge System, dated December 1, 1980.

Response:

The scram discharge system for WNP-2 has been evaluated against the Generic Safety Evaluation Report, "BWR Scram Discharge System," dated December 1, 1980. In short, the evaluation indicated that the WNP-2 scram discharge system needed upgrading in the following areas:

- Addition of redundant vent and drain isolation valves
- Addition of redundant and diverse level instrumentation for scram
- Relocation and repiping of instrument piping directly to the scram instrument volume
- Addition of new surveillance and operating procedures

A summary of our evaluation results is provided below:

FUNCTIONAL CRITERIA

The scram discharge volume (SDV) shall have sufficient capacity to receive and contain water exhausted by a full reactor scram without adversely affecting control rod drive scram performance.

WNP-2 compliance:

The WNP-2 SDV system is currently designed to meet the 3.34 gallons per drive requirement specified in the GE Design Specification 22A4260. This is an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

SAFETY CRITERIA

a. No single active failure of a component or service function shall prevent a reactor scram under the most degraded conditions that are operationally accepted.

WNP-2 Compliance:

The WNP-2 system has been designed to meet single failure criteria. The SDV is designed with an integral instrument volume (IV) which provides direct and immediate detection of liquid accumulation. The SDV instrumentation is redundant and single failure proof (including partial loss of service functions).

b. No single failure shall result in uncontrolled loss of reactor coolant.

WNP-2 Compliance:

A redundant air-operated vent valve and drain valve will be added on the SDV in series to insure system isolation during reactor scrams. This includes independent solenoid valves for each set of air-operated vent and drain valves.

c. The scram discharge system instrumentation shall be designed to provide redundancy, to operate reliably under all conditions, and shall not be adversely affected by hydrodynamic forces or flow characteristics.

WNP-2 Compliance:

Six additional diverse level sensors will be added to the SDV system to ensure diversity and redundancy in level monitoring and scram functions. Common cause failures will be considered in the selection of the instruments. This is in agreement with Alternative 3 of the "Acceptable Compliance" statement for this item in the Generic SER.

d. System operating conditions which are required for scram shall be continuously monitored.

WNP-2 Compliance:

The addition of the level switches described in (c) above and periodic surveillance testing of the instruments will provide a continuous means of monitoring the SDV liquid level and insuring instrument reliability. This is an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

e. Repair, replacement, adjustment, or surveillance of any system component shall not require the scram function to be bypassed.

WNP-2 Compliance:

During routine surveillance testing, instrument repair, or calibration, the associated logic will be placed in a half-scram (1 out of 2) configuration in accordance with the plant technical specifications.* This is an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

OPERATIONAL CRITERIA

a. Level instrumentation shall be designed to be maintained, tested, or calibrated during plant operation without causing a scram;

b. The system shall include sufficient supervisory instrumentation and alarms to permit surveillance of system operation;

c. The system shall be designed to minimize the exposure of operating personnel to radiation;

d. Vent paths shall be provided to assure adequate draining in preparation for scram reset;

e. Vent and drain functions shall not be adversely affected by other system interfaces. The objective of this requirement is to preclude water backup in the scram instrument volume which could cause spurious scram.

WNP-2 Compliance:

1. The system logic is designed as a one out of two, twice configuration. Each of the associated instrument channels is capable of being separately isolated for maintenance, testing, or calibration without inadvertently scramming the reactor.

2. The SDV is provided with a high liquid level alarm on each IV to alert the operator to liquid accumulation in the SDV.

3. The SDV system has been designed in accordance with GE Design Specification 22A4260 to minimize the exposure of operating personnel to radiation. In addition, the system is being reviewed as part of the WNP-2 ALARA program.

4. The SDV vents directly to the reactor building atmosphere and is independent from other plant vent systems.

5. The vent and drain system for the SDV is totally independent from other plant systems, and is therefore not susceptible to blockage or water buildup through system interfaces.

DESIGN CRITERIA

a. The scram discharge headers shall be sized in accordance with GE OER-54 and shall be hydraulically coupled to the instrumented volume(s) in a manner to permit operability of the scram level instrumentation prior to loss of system function. Each system shall be analyzed based on a plant-specific maximum inleakage to ensure that the system function is not lost prior to initiation of automatic scram. Maximum inleakage is the maximum flow rate through the scram discharge line without control rod motion summed over all control rods. The analysis should show no need for vents or drains.

WNP-2 Compliance:

The WNP-2 IVs have been designed as vertical extensions attached directly to the SDV. This configuration provides a direct hydraulic couple between the SDV and IVs and insures continuous liquid level monitoring in the SDV. This is an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

The high level scram setpoint and the SDV/SDIV system capacity ensure that scram capability is maintained even in the event of maximum inleakage into the SDV prior to a scram. Analysis, assuming the maximum inleakage of 5 gpm and using the actual calculated piston-over area to determine the scram volume requirements, shows that adequate scram discharge volume will remain in the SDV system at the time that a scram is initiated.

b. Level instrumentation shall be provided for automatic scram initiation while sufficient volume exists in the scram discharge volume.

WNP-2 Compliance:

The WNP-2 SDV is adequately coupled to the IV to allow proper instrument operation. The SDV instrument setpoint for scram was established to insure an available volume of 3.34 gallons per drive (185 drives). This is an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

c. Instrumentation taps shall be provided on the vertical instrument volume and not on the connected piping.

WNP-2 Compliance:

All the WNP-2 SDV instrumentation will be relocated and repiped directly to the IV instead of the vent and drain piping. Procedures will be modified to include functional testing of SDV level instrumentation after each scram. This is an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

d. The scram instrumentation shall be capable of detecting water accumulation in the instrumented volume(s) assuming a single active failure in the instrumentation system or the plugging of an instrument line.

WNP-2 Compliance:

The addition of the redundant and diverse instruments described under Safety Criterion 3 and rerouting of the instrument piping to the IV provide an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

e. Structural and component design shall consider loads and conditions, including those due to fluid dynamics, thermal expansion, internal pressure, seismic considerations, and adverse environments.

WNP-2 Compliance:

The WNP-2 SDV design is in compliance with the latest GE design criteria as outlined in GE Design Specification 22A4260. In addition, the system will be reviewed as part of the equipment qualification program.

f. The power-operated vent and drain valves shall close under loss of air and/or electric power. Valve position indication shall be provided in the control room.

WNP-2 Compliance:

The WNP-2 present design configuration meets these requirements.

g. Any reductions in the system piping flow path shall be analyzed to assure system reliability and operability under all modes of operation.

WNP-2 Compliance:

The WNP-2 SDV header system is designed as a continually expanding path from the 185 3/4" individual scram discharge (withdrawal) lines to one of two integrated SDV/IV systems (one system per approximately half the drives). Each integrated SDV/IV system consists of a continuously downsloping piping run expanding from the SDV (consisting of seven 6" return headers from the individual hydraulic control unit (HCU) banks to an 8" combined return header) to the 12" vertically oriented IV. The location where blockage need be assumed (piping less than 2" diameter) is in the 3/4" discharge line from the individual HCU. Blockage here would only cause failure of one control rod to insert. This is an acceptable consequence for a single failure and has been evaluated as part of the plant design basis. Accordingly, this design complies with the "Acceptable Compliance" statement for this item in the Generic SER.

h. System piping geometry (i.e. , pitch, line size, orientation) shall be such that the system drains continuously during normal plant operation.

WNP-2 Compliance:

The WNP-2 SDV has been designed to insure a positive downward slope of scram header and drain piping.

i. Instrumentation shall be provided to aid the operator in the detection of water accumulation in the instrumented volume(s) prior to scram initiation.

WNP-2 Compliance:

Each IV is provided with high liquid level and rod block instrumentation attached directly to it. The generic SER states that this is acceptable.

j. Vent and drain line valves shall be provided to contain the scram discharge water, with a single active failure and to minimize operational exposure.

WNP-2 Compliance:

As stated under Safety Criterion 2, redundant airoperated vent and drain valves will be provided for system isolation. This is an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER.

SURVEILLANCE CRITERIA

a. Vent and drain valves shall be periodically tested.

WNP-2 Compliance:

The vent and drain valves will be tested in accordance with the plant Technical Specifications to verify valve closure in less than 30 seconds (current GE specification). This is an acceptable means of compliance as documented in the "Acceptable Compliance" statement for this item in the Generic SER. Technical Specification 4.1.3.1.4 documents this requirement.

b. Verifying and level detection instrumentation shall be periodically tested in place.

WNP-2 Compliance:

The SDV instrumentation will be tested in accordance with the plant Technical Specification which will include post scram testing to verify instrument operability. The testing will be performed in accordance with 3/4.3.6 and 3/4.3.1 of the Technical Specification.

c. The operability of the entire system as an integrated whole shall be demonstrated periodically and during each operating cycle, by demonstrating scram instrument response and valve function at pressure and temperature at approximately 50% control rod density.

Surveillance testing will be performed in accordance wih the plant Technical Specification 4.1.3.1.4.

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Q. 010.042
(4.6)
RSP

Demonstrate that a slow or partial loss of air pressure to the scram discharge valves will not result in the following:

a. Rapid filling of both the scram discharge volume and the instrument volume due to the lifting of most or all scram discharge valves, with consequent loss of adequate scram discharge volume.

b. Loss of reactor coolant due to the combination of lifting of most or all scram discharge valves, without compensating closure of the vent and drain valves, with consequent environmental effects inside containment.

Unless it can be demonstrated that no adverse effects can result, a system shall be provided and described in this section to protect against these two conditins.

Response:

a. The high level scram setpoint and the SDV/SDIV system capacity ensure that scram capability is maintained even in the event of maximum inleakage into the SDV prior to a scram. Analysis, assuming the maximum inleakage of 5 gpm and using the actual calculated piston-over area to determine the scram volume requirements, shows that adequate scram discharge volume will remain in the SDV system at the time that a scram is initiated.

b. The partial loss of air pressure does not result in the uncontrolled release of reactor coolant to the reactor building should all or most of the scram discharge valves lift. When the water buildup reaches scram initiation level in the IV, a scram signal is produced. This will cause the air supply to the vent and drain valves to vent, thereby ensuring that the vent and drain valves close and isolate. For leakage rates which do not result in

buildup in the IV, the leak will drain to the reactor building equipment drain system. The drain system will alarm for leakage rates greater than five (5) gallons per minute. The operator can then take appropriate action, e.g., isolate the leak, scram the reactor, increase air pressure, etc., as required.

See response to Question 010.041 for additional information.

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Q. 010.043
(4.6)

Describe the effects on the safety and operability of the control rod drive hydraulic system if the following control rod drive system valves either fail closed or fail open:

a. Drive water pressure control valve (between F060 and F061);

b. Cooling water pressure control valve (between F070 and F071).

Response:

The function of the F003 pressure control valve (PCV) is to provide a means of adjusting the drive water header and cooling water header pressures. The F003 PCV is a manually controlled motor-operated valve which is controllable from the main control room. Indicating lights are provided in the control room for the valve full open and full closed positions. Adjustment of the F003 PCV in concert with adjustments to the F002 flow control valve permit adjustment of the drive water header pressure to approximately 260 psi above vessel pressure while at the same time, maintaining the drive cooling water header pressure at approximately 20 psi above vessel pressure.

If the F003 PCV were to fail to a full open position, the cooling water pressure would increase and the drive water pressure would decrease. The resulting cooling water pressure increase could cause control rods to drift inward. The existence of rod drifts would be alarmed to the control room operator for appropriate action. The resulting drop in drive water pressure would make normal control rod notch movements impossible but would not affect the ability of the scram function.

Conversely, if the F003 PCV were to fail to a full closed position, the cooling water pressure would decrease while the drive water pressure would increase. The reduction in cooling water pressure (and flow) would eventually lead to high CRD temperatures being alarmed in the control room. The CRD system's scram function would not be affected by the increase in drive water pressure. In the limiting case, the resulting increase in drive water pressure would reach up to the shut-off pressure of the supply pump (1750 psig). The occurrence of this condition during withdrawal of a drive at zero reactor pressure will result in a drive pressure increase from 260 psig to no more than 1750 psig. Calculations and tests

indicate that the drive would accelerate from 3 inches per second to no more than 6.5 inches per second. The rod movement would stop after the driving signal is removed or rod block is enforced by the reactor manual control system (RMCS). In the unlikely event where RMCS fails to enforce a rod block, the peak fuel enthalpy for drive speeds of 6.5 inches per second is well below the fuel cladding failure threshold design limit. Therefore, due to provisions in the system design and margin in the fuel design, this postulated scenario will not compromise the integrity of the fuel.

In both of the cases described above, the manually operated bypass PCV (F004) in conjunction with the isolation gate valves located upstream and downstream of the F003 PCV would enable the operators to take corrective action.

In conclusion, although the failure to the full open or full closed position of the drive/cooling water PCV will cause perturbation in the CRD system operation, it does not present a safety problem to affect the scram capability of the CRD system.

The PCV F005 was deleted from the CRD hydraulic system in the process of implementing the CRD return line deletion modifications, therefore, this question is not applicable.

See the response to Question 211.130 for additional details on the deletion of the CRD return line.

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Q. 010.044
(4.6.1.1.2.4)

Table 1.3-8 indicates specific design changes from the PSAR to the FSAR for the CRD system. The design changes for the CRD return line modification addressed in Question 211.019, have not been included in the text description of the FSAR and Figures 4.6-5a, 4.6-5b, and 4.6-6a have not been revised. Revise the text description in the FSAR to reflect the specific design changes in Table 1.3-8 for the CRD system and modify the above figures accordingly.

Response:

Figures 4.6-5a, 4.6-5b, 4.6-6a, 4.6-6b, and 4.6-6c are revised to reflect the CRD return line modification. Figures 4.6-6d and 4.6-6e are deleted. Also, text 4.6.1.1.2.4, 4.6.1.1.2.4.1, 4.6.1.1.2.4.2.3, 4.6.1.1.2.4.2.4, and 4.6.2.3.2.2.8 are revised accordingly. Section 4.6.1.1.2.4.2.5 is deleted.

Q. 010.045
(4.6.1.1.2.4.1)

The scram discharge volume header piping is sized to receive and contain all water discharged by the control rod drives during a scram, independent of the instrument volume. Show quantitatively how a minimum volume of 3.34 gallons per drive is required since approximately 4 gpm is required to insert the rods with up to an additional 0.34 gpm required for cooling.

Response:

The 3.34 gallon minimum volume requirement is in no way related to the 4 gpm drive insert flow rate or the 0.34 gpm cooling water flow rate. The nominal 4 gpm drive insert flow is the volumetric flow rate delivered to the underside of the control rod drive (CRD) piston to displace the piston in the upward direction and achieve normal rod insertion at about 3 inches per second. The 0.34 gpm cooling water is also delivered to the underside of the drive piston and from here it is discharged to the reactor pressure vessel (RPV) via an engineered flow path up through the annular thermal sleeve region between the CRD piston mechanism and the CRD housing. On the other hand, the scram discharge volume is sized to provide a low pressure discharge point for the volume of water above the drive piston displaced during the period of scram insertion and an additional, conservatively defined, maximum leakage from the RPV to the top of the drive piston during the scram. The volume of water over the drive piston of a fully withdrawn CRD is 0.76 gallons. To this is added a conservative 10 gpm leakage flow from the RPV for an extended period of 10 seconds. (Normal full rod insertion is complete in less than 3 seconds.) Finally, some more volume is added to accommodate the air potentially trapped in the SDV so as to assure that the SDV pressure at 10 seconds after the time of scram initiation is $\leq$ 65 psig. The sum of the 0.76 gallons displaced from the top of the drive piston, the 10 seconds of 10 gpm post-scram leakage flow from the RPV and the free volume required for the air trapped in the SDV adds up to the specified minimum value of 3.34 gallons per CRD.

Q. 010.046
(4.6.1.1.2.4.2.2)

In Figure 4.6-5b and Drawing M528, pressure transmitter (N005) transmits a signal to a pressure switch (N600) in the process instrumentation panel in the control room, which energizes an annunciator in the control room at any time pressure in the charging header falls below the setpoint. Explain why an alarm on high is indicated for the pressure switch (N600) instead of an alarm on low which would provide protection against charging header pressure falling below the setpoint.

Response:

The charging water header of the control rod drive (CRD) is monitored for high pressure since high charging water header pressure indicates the existence of an abnormal condition in the CRD hydraulic system (e.g., such as a failed close flow control valve). The pressure indicating switch on the charging water header (C11-N600) is set to actuate the control room annunciator if the charging water header pressure exceeds a nominal 1510 psig setpoint (the alarm is actuated on an increasing pressure). Neither sustained high charging water pressure nor CRD drive water pump operation is required to successfully scram the plant. Each of the control rod drives has its own hydraulic control unit (HCU) which operates independently of any others. Scram is achieved on either HCU accumulator pressure or a combination of accumulator pressure and reactor pressure. Each HCU is safety grade and has its own accumulator. The condition of the accumulators is continuously monitored by the reactor manual control system. Loss of pressure and/or leakage from any of the accumulators is detected by PSL-130 and LDS-129, respectively, for each accumulator, as shown in Figure 4.6-5b. Both occurrences are annunciated and a light signal identifies the particular scram accumulator. This instrumentation, existing locally at each HCU, provides the necessary indication of accumulator charge pressure irrespective of the pressure in the nonessential charging water header.

Q. 010.047
(4.6.1.1.2.4.2.4)

Revised FSAR Figure 4.6-5b (Amendment 16) does not appear to show valves F129, F130, F131, and F132. The applicant should explain if this is what is meant in the response by "removing the discrepancy identified above". In addition, the revised FSAR page was not provided as stated in the response. The applicant should provide the revised FSAR page and describe the revision.

Response:

FSAR Figure 4.6-5B (Amendment 16) does not show valves F129, F130, F131, and F132 because these valves were removed from the CRD system when the CRD return line to the RPV was eliminated.

What is meant by "removing the discrepancy identified above" in the response to Question 211.134 (later changed to 010.047) is that by submitting revised FSAR Figure 4.6-5b, the discrepancy identified in Question 010.047 between the FSAR text, M528 (FSAR 3.2-4) and FSAR Figure 4.6-5b was resolved.

The words "revised FSAR page change attached" referred to FSAR Figure 4.6-5b.

Q. 010.048
(4.6.1.1.2.4.3.9)

The text description of the scram accumulator indicates that a check valve in the accumulator charging line prevents loss of water pressure in the event supply pressure is lost. The symbol for valve 111 in Figure 4.6-5b and Drawing M528 appears to be that of a normally open globe valve instead of a stop-check globe valve. Explain this apparent discrepancy.

Response:

There is no disagreement between 4.6.1.1.2.4.3.9 and Figure 4.6-5b and Drawing M528 concerning valve 111. "A check valve in the accumulator charging line prevents loss of water pressure in the event the supply pressure is lost" refers to valve 115 and to the "charging water". Valve 111 is closed only when the pressure instrumentation is being serviced and when the nitrogen charging station is being connected and disconnected.

Q. 010.049
(5.2.5)
(7.6.2)
(12.3.4)

Provide a detailed discussion of the sensitivity and response times of the containment airborne radiation monitoring systems for a number of containment background activity levels. The background activity levels which should be considered are those levels in the containment that would result from leakage through the RCPB assuming: (1) relatively clean water in the reactor coolant system at the initial operation of the WNP-2 facility at power; and (2) the maximum level of activity in the reactor coolant permitted by the WNP-2 Technical Specifications. In responding to this item, assume both the normal and the maximum leakage rates identified in your response to Question 211.005. Indicate your assumptions in estimating the response times of the containment airborne radiation monitoring systems (e.g., the preset alarm level for higher background leakage and the plateout factor).

Response:

The two types of radiation monitors used in the WPPSS Nuclear Project No. 2, for monitoring the drywell atmosphere, are the particulate monitor and the noble gas monitor. The sensitivity of these two types of monitors is given in Figure 010.049-1 and 010.049-2. The same detector is used in both monitors. The detector's noise level is about 25 cpm. The minimum detectable concentration is based on doubling the background count rate. The count ratemeter range and the minimum sensitivity of both types of monitors are:

Noble Gas

Count Ratemeter Range:

1.4×10^{-7} μci/cc - 1.4×10^{-1} μci/cc for Kr-85

Minimum Detectable Concentration:

3.6×10^{-7} μci/cc for Kr-85

Particulate

Ratemeter Range:

2.9×10^{-12} μci/cc - 2.9×10^{-6} μci/cc for Sr/Y-90

Minimum Detectable Concentration:

7.4×10^{-12} μci/cc for Sr/Y-90

The quantity of drywell atmosphere that flows through the filter of the particulate monitor and then through the 2.2 liters chamber of the noble gas monitor is 3 cfm. The drywell atmosphere sample is returned to the drywell. (The free volume of the drywell is 200,540 ft^3). There is a charcoal filter after the particulate and before the noble gas detector. The filter efficiency of the particulate filter is assumed to be 100 percent. Similarly, the efficiency of the charcoal filter is assumed to be 100 percent for all halogens.

The 2-inch diameter scintillation crystal, of the particulate detector, is 1/4-inch away from the face of the filter tape. The filter tape is 2.5 inches wide and moves at 1"/hr.

As per the response to Question 211.005, the total minimum identifiable and unidentifiable leakage rate is taken to be 2.1 gpm. The total maximum identifiable and unidentifiable leakage rate is taken to be 5.5 gpm. The leakage is measured at drywell environmental conditions and thus the water density is assumed to be 1g/cc. Based upon potential sources of leakage (such as number and nominal size of valves), it is estimated that 38.5 percent of the leakage is due to steam leakage. Collection of the identifiable leakage is not seal-tight and thus, volatile radioisotopes can escape into the drywell atmosphere. Water leakage is assumed to be flashing, and that there is an instantaneous mixing of the volatile radioisotopes with the drywell atmosphere. It is assumed that the noble gases are in the steam phase and the small quantities of noble gases in the liquid phase are neglected. On the other hand, it is assumed that the quantities of particulate radioisotopes in the steam phase are small and thus, are neglected.

Decay was considered for the radioisotopes in the drywell and for those radioisotopes accumulated on the particulate filter. No decay was considered, however, for the radioisotopes while in transit to the detectors and while in the noble gas detector chamber (for the atmosphere in the chamber, exchange rate is 1.55 seconds). It is assumed that plating, settling,

impingement, etc., redue the specific concentration of particulate isotopes in the drywell atmosphere by a factor of 1000 before the air flow reaches the filter of the paticulate monitor.

In order to be responsive to the question, the source concentrations used in this analysis were taken to be those in Table 5 of ANS/ANSI N237-1976, reduced by 1/100, as representative of "relative clean water in the reactor coolant". The design basis concentrations, as per General Electric specification document No. 22A2703F, Revision 3, were used as representative of the maximum expected level of radioactivity within the reactor coolant.

The criterion used, as an indication of leakage increase, is the doubling of the background count rate within one hour for 1 gpm (additional) leakage. Each detector was evaluated with respsect to responding to the criterion. See Table 010.049-1 for the results of the analysis.

In summary, the particulate monitor meets the requirements stated in Regulatory Guide 1.45 for the minimum activity concentration of radioisotopes in the reactor coolant. For the cases where it is assumed that the design basis activity exists in the reactor coolant, the background activity exceeds the particulate monitor range. However, the detector can be desensitized accordingly. It can be shown that if a reactor coolant activity is selected based upon the guidance contained in Regulatory Guide 1.45; i.e., if "a realistic primary coolant radioactivity concentration" is used, e.g., equal to that given in Table 5 in ANS/ANSI N237-1976, and expanding the criterion of double background count rate to the desensitized monitor, the requirements stated in the regulatory guide will be met.

The noble gas monitor, however, even though its sensitivity is consistent with Regulatory Guide 1.45 requirements, would not be capable of detecting the additional 1 gpm leakage within one hour utilizing the criterion of double background count as positive indication. The noble gas monitor does, however, provide the most reliable and fastest means of ascertaining increased activity within containment with unidentified leakages higher than 1 gpm.

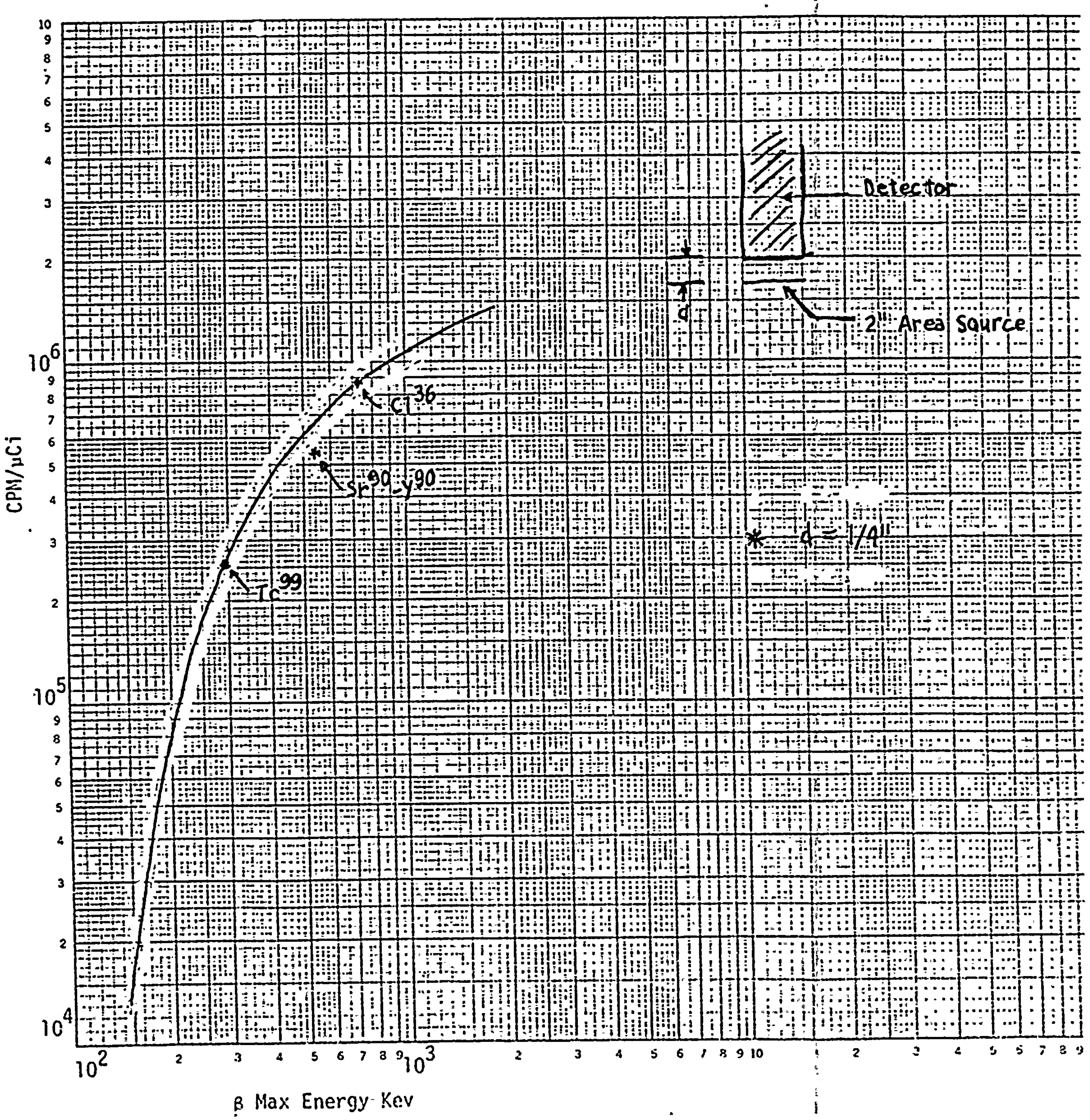
TABLE 010.049-1

RESULTS OF MONITOR ANALYSIS

CASE	RESULTS OF ANALYSIS FOR NOBLE GAS MONITOR	RESULTS OF ANALYSIS FOR PARTICULATE MONITOR
a. Minimum activity in reactor coolant b. Minimum background leak rate c. Duration of background leakage = one day d. Unidentified leakage of 1 gpm is introduced.	When the background of about 5 cpm is added to the detector's noise level of $\sim$25 cpm, the total cpm by the detector before the event is about 30. With the increase of 1 gpm leakage, the count rate would increase one hour after the event to only about 32 cpm.	The background count rate would double as a result of 1 gpm unidentified leakage within about 46 minutes.
a. Minimum activity in reactor coolant b. Maximum background leak rate c. Duration of background leakage = one day d. Unidentified leakage of 1 gpm is introduced.	When the background of about 12 cpm is added to the detector's noise level of $\sim$25 cpm, the total cpm by the detector before the event is about 37. With the increase of 1 gpm leakage, the count rate would increase, one hour after the event, to only 39 cpm.	The background count rate would double as a result of 1 gpm unidentified leakage within about 49 minutes.
a. Maximum activity in reactor coolant b. Minimum background leak rate c. Duration of background leakage = one day d. Unidentified leakage of 1 gpm is introduced.	When the background of about 81 cpm is added to the detector's noise level of $\sim$25 cpm, the total cpm by the detector, before the event, is about 106. With the increase of 1 gpm leakage, the count rate would increase, one hour after the event, to only about 136 cpm.	The background count rate exceeds the detector's sensitivity. Analysis shows that if the detector's sensitivity were high enough, it would take about 51 minutes for the background count rate to double.
a. Maximum activity in reactor coolant b. Maximum background leak rate c. Duration of background leakage = one day d. Unidentified leakage of 1 gpm is introduced.	When the background of about 213 cpm is added to the detector's noise level of $\sim$25 cpm, the total cpm by the detector, before the event, is about 238. With the increase of 1 gpm leakage, the count rate would increase, one hour after the event, to only about 269 cpm.	The background count rate exceeds the detector's sensitivity. Analysis shows that if the detector's sensitivity were high enough, it would take about 53 minutes for the background count rate to double.

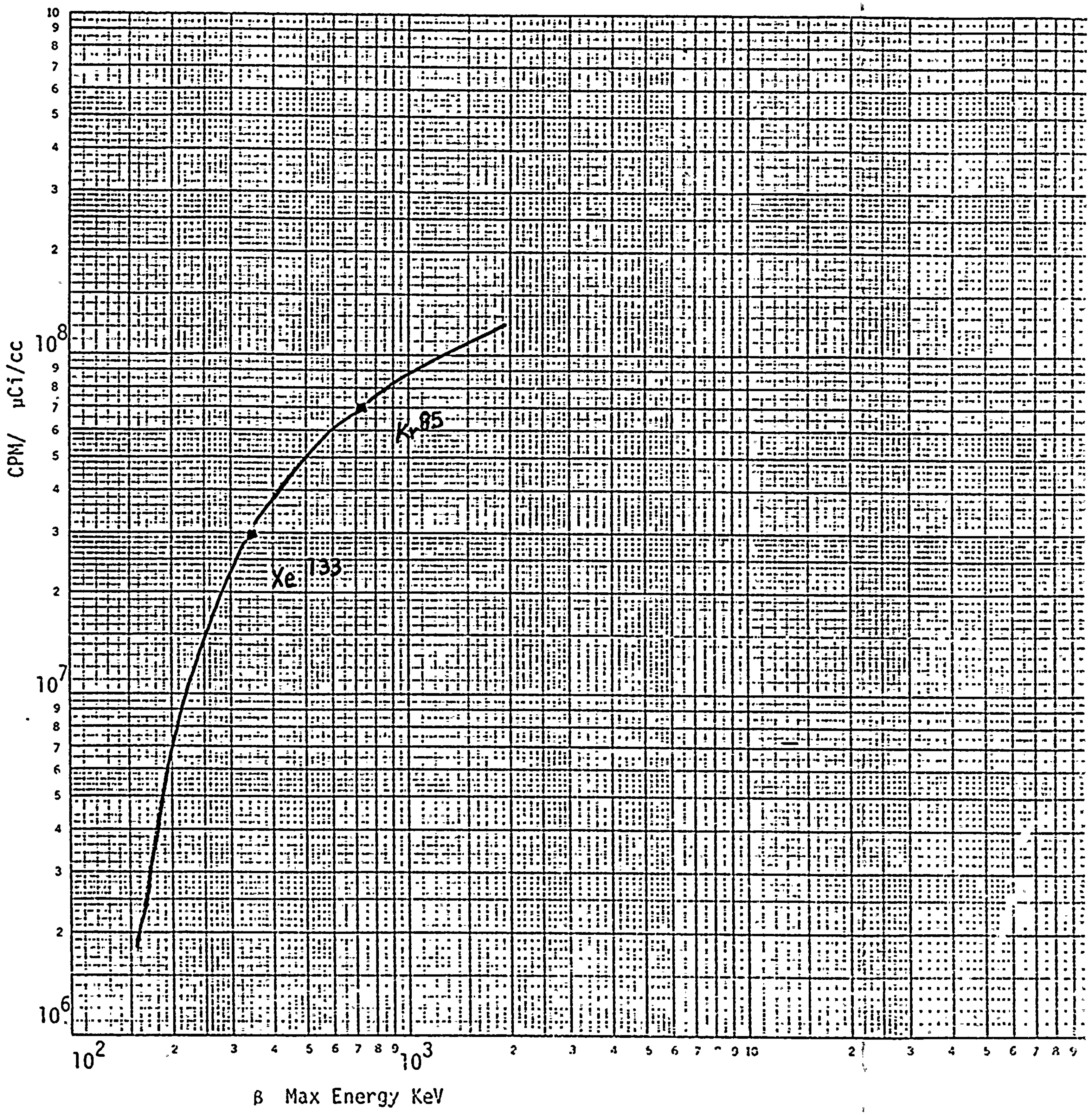
TABLE 010.049-1 (Continued)

CASE	RESULTS OF ANALYSIS FOR NOBLE GAS MONITOR	RESULTS OF ANALYSIS FOR PARTICULATE MONITOR
a. Minimum activity in reactor coolant b. Minimum background leak rate c. Duration of background leakage = 100 days d. Unidentified leakage of 1 gpm is introduced.	When the background of about 10 cpm is added to the detector's noise level of ∼25 cpm, the total cpm by the detector, before the event, is about 35. With the increase of 1 gpm leakage, the count rate would increase, one hour after the event, to only about 36 cpm.	The background count rate will double as a result of 1 gpm unidentified leakage, within abut 51 minutes.
a. Minimum activity in reactor coolant b. Maximum background leak rate c. Duration of background leakage = 100 days d. Unidentified leakage of 1 gpm is introduced.	When the background of about 25 cpm is added to the detector's noise level of ∼25 cpm, the total cpm by the detector, before the event, is about 50. With the increase of 1 gpm leakage, the count rate would increase, one hour after the event, to only about 52 cpm.	The background count rate would double as a result of the 1 gpm unidentified leakage, within about 53 minutes.
a. Maximum activity in reactor coolant b. Minimum background leak rate c. Duraton of background leakage = 100 days d. Unidentified leakage of 1 gpm is introduced.	When the background of about 165 cpm is added to the detector's noise level of ∼25 cpm, the total cpm by the detector, before the event, is about 190. With the increase of 1 gpm leakage, the count rate would increase, one hour after the event, to only about 219 cpm.	The background count rate exceeds the detector's sensitivity. Analysis shows that if the detector's sensitivity were high enough, it would take about 57 minutes for the background count rate to double.
a. Maximum activity in reactor coolant b. Maximum background leak rate c. Duration of background leakage = 100 days d. Unidentified leakage of 1 gpm is introduced.	When the background of about 431 cpm is added to the detector's noise level of ∼25 cpm, the total cpm by the detector, before the event, is about 456. With the increase of 1 gpm leakage, the count rate would increase, one hour after the event, to only about 485 cpm.	The background count rate exceeds the detector's sensitivity. Analysis shows that if the detector's sensitivity were high enough, it would take about 58 minutes for the background count rate to double.



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	PARTICULATE SYSTEM EFFICIENCY	FIGURE 010. 049-1

AMENDMENT NO. 21
December 1981



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	GASEOUS EFFLUENT SYSTEM EFFICIENCY	FIGURE 010. 049-2

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Q. 010.050
(5.2.5)

The response to Question 211.007 requires additional information. It is unclear how the comparison will be made between the radioactivity monitoring and the sump level monitoring.

Describe briefly the mechanics of making these data comparisons. What calibration and operability verification tests will be performed for each independent leakage detection system? Which leakage detection system is to be used as the reference for comparison with the other systems? Do the radiation monitoring systems have radioactive sources (check sources) built into the systems?

Response:

Radiation monitors are useful as leak detection devices because of their sensitivity and rapid response to leaks. After several weeks of full power operation, a set level of background radiation will be established. Any sudden or unexplained increase in background radiation would indicate a possible primary coolant leak within the primary containment. If an increase is noted, a comparison with other leak detection devices having a relationship to each other will be made, particularly the equipment and floor drain flow rate monitors, and the reactor building sump pumps activation on high sump level. Using the flow rate monitors as a reference, the comparisons provide independent indications of a leak within the primary containment.

The radiation monitoring channels are redundant, allowing cross-checks between channels. Each channel is equipped with test sources and purge systems so that proper operation of each individual channel can be verified from the detector through to the indicator.

A discussion of calibration and operability verification tests for the leak detection system is discussed in 7.6.2.4.a.

Q. 010.051
(5.2.5)

Identified leakage is determined during preoperational testing or is measurable during reactor operation. Will the identified leakage be measured regularly and recorded? If so, provide the frequency that these data will be recorded and indicate what procedural guidelines are to be used to change the magnitude of the base identified leakage rate?

Response:

The identified leakage is measured continuously and the leakage rate will be calculated and recorded on a frequency of at least once per 12 hours in accordance with the plant technical specifications. The procedures describing how the identified leakage rate is determined will include provisions for showing the identified leakage rate has not exceeded the maximum allowable value of 25 gpm, including no more than 5 gpm unidentified leakage.

Each equipment leak-off connection has been provided with a temperature element which will identify to the operator that a higher than normal temperature exists at that particular location.

Q. 010.052
(5.2.5)

Standard Review Plan 5.2.5 specifies that unidentified leakage should be collected separately from the identified leakage so that a small unacceptable unidentified leak is not masked by larger acceptable identified leakage. Section 5.2.5 of the FSAR does not clearly indicate that separate collection of identified and unidentified leakage is provided.

Provide assurances that identified and unidentified leakage will be collected separately. If separate collection is not to be provided, provide justification for use of a common collection reservois and show that a small unidentified leak of about 1 gpm would be recognized within one hour.

Response:

Identified and unidentified leakage are collected separately. Identified leakage is collected, monitored, and indicated by the equipment drain system (see Figure 3.2-9) while the unidentified leakage is collected, monitored, and indicated by the floor drain system (see Figure 3.2-10). Section 5.2.5.6 refers, in part, to 7.6.1.3 for further explanation. In 7.6.1.3.4, 7.6.1.3.5 and 7.6.1.3.6, the two separate collection systems are described.

Q. 010.053
(5.2.5)

Provide a list of all indications available to the control room operator for evaluating and detecting unidentifed leakage. Show how the operator will determine the amount of leakage by observing the indications that are available to him, including the need for unit conversion (count rate to gpm, etc.). If the monitoring is computerized, discuss the backup procedures available should the computer become inoperative.

Response:

The following indications are available to the control room operator for evaluating and detecting unidentified leakage:

Drywell Pressure Records	-3 to +3 psig 0 to 25 psig 0 to 180 psig
Drywell Temperature Recorders	50° to 400°F
Drywell Floor Drain Total Flow Recorder	0 to 30 gpm
Reactor Building Floor Drain Sump Fillup Rate Timer	0 to 150 Min.
Reactor Building Floor Drain Sump Pump Out Rate Timer	0 to 30 Min.
Drywell Cooler Cooling Water Differential Temperature Recorder	0 to 150°F
Reactor Vessel Water Level Recorders	-353.2" to -153.2" -185" to +25" -35" to +85"
Drywell Atmosphere Radiation Monitors	1 to 10^6 CPS

The indications listed above have no definitive correlation between their engineering units. For example, a specific count rate indicated by the drywell atmosphere radiation monitors cannot be directly converted to a leak rate in gpm, nor can a high drywell temperature be converted to an equivalent gpm leak rate. The indications listed are provided as an early warning to the operator of a potential leak. The actual unidentified leak rate is determined by observing the drywell floor drain system flow rate on flow recorders provided in the control room. The monitoring is not computerized.

Q. 010.054
(9.1.1)

Provide the K_{eff} and the density for optimum moderation for the new fuel storage facility, assuming the infinite array of maximum enriched new fuel for the optimum case. Describe the preventive measures taken to assure that $K_{eff} \leq 0.98$ for the new fuel storage facility for all moderating conditions. Alternately, demonstrate that no moderating condition between 100% water and 0% water densities can credibly exist.

Response:

Incorporated into Section 9.1.1

Q. 010.055
(9.1.2)

Describe, discuss, and verify that the maximum potential kinetic energy contained in all objects of less weight than a spent fuel assembly which will be handled over spent fuel will not exceed the effects of the fuel handling accident described in Section 15.7.4 of the FSAR.

Response:

Tables 010.055-1, 010.055-2, and revised 9.1.2.3.2 have been prepared as a response to this question. This response also closes out an open item from the Auxiliary Systems Branch meeting October 7, 1981.

TABLE 010.055-1

LIGHT LOADS OVER THE SPENT FUEL POOL

	Item	Distance Above Pool Surface or Above Fuel Rack in Pool (in feet) Above Pool	Above Rack	Weight No.	Kinetic Energy at Impact (top of rack) ft/lb
1.	Channel Bolt Wrench	4		40	982
2.	Channel Handling Tool	14		75	2,592
3.	Channel Gauging Fixture		13*	210	2,389
4.	General Purpose Grapple	14		25	864
5.	Clam Shell Retriever	4		14	344
6.	Manipulator Grapple	4		50	1,228
7.	Actuating Pole	3#	20.5*	100	2,075
8.	General Area Underwater Light	4		40	982
9.	In Core Detector Cutter	4		150	3,684
10.	Fuel Support Grapple	4		147	3,610
11.	Peripheral Orifice Grapple	4		45	1,105
12.	Peripheral Orifice Holder	4		130	3,193
13.	Blade Guide	3*	20.5*	170	3,527
14.	Fuel Bail Cleaner	4		100	2,456
15.	Grid Guide	3#	20.5*	175	3,631

* Distance of CG to top of rack.
\# Distance of CG above pool.

TABLE 010.055-1 (Continued)

Item	Distance Above Pool Surface or Above Fuel Rack in Pool (in feet) Above Pool	Above Rack	Weight No.	Kinetic Energy at Impact (top of rack) ft/lb
16. Dummy Fuel Assembly		8.2	600	4,305
17. Peripheral Fuel Support Plug	4		300	7,368
18 Fuel Grapple		23.5	100	2,056
19. Control Tube Grapple	4		45	1,105
20. Guide Tube Grapple	4		35	860
21. Control Rod Latch Tool	4		45	1,105
22. Fuel Bundle Sampler	4		650	15,964
23. Fuel Bundle & Channel		8.2	697	5,001
24. Fuel Bundle & Channel w/Grapple(1)		8.2/15.7	697/100	6,375

* Distance of CG to top of rack.
\# Distance of CG above pool.

Note (1): Assumed to be fuel grapple.

TABLE 010.055-2

LIGHT LOADS OVER REACTOR VESSEL CORE

Item	Weight of Item	Height In Air	Potential Energy In Air Ft./Lb.	.875 x Height In Water	Potential Energy Water Ft./Lb.	Total Potential
General Purpose Grapple	25#	6'	150	46'	1,150	1,300
Manipulator Grapple	50	6	300	46	2,300	2,600
J-Hook or L-Hook with 6 Sections of Pole + 5 Connectors & 1 Tee Handle	47	6	282	46	2,162	2,444
Rail Clamp	3	29	87	27	81	168
Clam Shell Retriever	15	6	90	46	690	780
Magnetic Retriever	2	6	12	46	92	104
General Area Underwater Light	40	29	1,160	27	1,080	2,240
Local Area Underwater Light	20	29	580	27	540	1,120
Drop Light	25	29	725	27	675	1,400
Underwater TV	25	29	725	27	675	1,400
Viewing Aid	11	6	66	46	506	572
Light Support Bracket	70	29	2,030	27	1,890	3,920
Fuel Support Grapple	87	6	522	46	4,002	4,524
Instrument Strongback	600	29	17,400	27	16,200	33,600
Peripheral Orifice Grapple	45	6	270	46	2,070	2,340
CRD Guide Tube Seal	150	29	4,350	27	4,050	8,400
In Core Guide Tube Seal	120	29	3,480	27	3,240	6,720
Peripheral Orifice Holder	130	29	3,770	27	3,510	7,280
Blade Guide	170	6	1,020	46	7,820	8,840
Fuel Ball Cleaner	100	6	600	46	4,600	5,200

TABLE 010.055-2 (Continued)

Item	Weight of Item	Height In Air	Potential Energy In Air Ft./Lb.	.875 x Height In Water	Potential Energy Water Ft./Lb.	Total Potential
Grid Guide	175	6	1,050	46	8,050	9,100
Dummy Fual Assembly	600	6	3,600	46	27,600	31,200
Fuel Grapple	1,000	6	6,000	46	46,000	52,000
In Vessel Storage Rack	575	29	16,675	27	15,525	32,200
Control Rod Grapple	45	29	1,305	27	1,215	2,520
CRD Guide Tube Grapple	35	29	1,015	27	945	1,960
Stud Handling Tool	135	29	3,915	27	3,645	7,560
RPV Stud	480	29	13,920	27	12,960	26,880
Shroud Head Butt Wrench	110	29	3,190	27	2,970	6,160
Head Stud Rack	300	29	8,700	27	8,100	16,800
Steamline Plug & Installing Tool	500	29	14,500	27	13,500	28,000

Q. 010.056
(9.1.3)

Your response to Question 010.021 is completely inadequate. Your design should be modified to provide one of the following alternatives:

a. A Seismic Category I, Quality Group C, tornado missile protected spent fuel pool cooling system including the secondary fuel pool heat exchanger cooling system.

b. A Seismic Category I, Quality Group C, tornado missile protected, makeup water supply to the spent fuel pool and HVAC (the HVAC design environment should be 212°F and 100% humidity). The structure above the refueling floor should be Seismic Category I and tornado missile protected.

c. A Seismic Category I, Quality Group C, makeup water supply to the spent fuel pool and the results of an analysis which verifies that with the loss of the structure above the refueling floor, cooling with only the Seismic Category I makeup, the most unfavorable atmospheric diffusion conditions (X/Q) that the site boundary dose will not exceed 25% of the limits specified in Title 10, Code of Federal Regulations, Part 100.

Response:

A Seismic Category I, Quality Group C, tornado missile protected spent fuel pool cooling system has been provided which satisfies Alternative a. See revised 9.1.3.

The valves, piping, and components of the cooling portion of the fuel pool cooling and cleanup system are located within the reactor building and are protected from tornado generated missiles. The cooling portion of the system is Seismic Category I, and will automatically isolate from the non-seismic cleanup portion of the system on low fuel pool water level. Remote-manual startup from the main control room of redundant active components of the cooling portion of the system is provided. Safety grade cooling water to the heat exchangers and fuel pool makeup water is available from the standby service water (SW) system and can be initiated by remote-manual operation from the main control room. To preclude leakage of service water into the spent fuel pool during operation of the Service Water system or leakage of

fuel pool cooling water into the Service Water system when the Service Water is not operating, the manual valves (SW-V-75AA and 75BB) and the motor operated valves (SW-V-75A and 75B) are kept normally closed when spent fuel pool temperatures are below 138°F. If during normal plant operations the spent fuel pool temperatures rises above 138°F, the manual valves will be maintained open. The manual valves are located on the west side of the 522 foot elevation of the reactor building and are accessible and can be opened if necessary following a LOCA prior to spent fuel pool temperatures exceeding 155°F. The access route to these manual valves is shown in Appendix J. Once the manual valve(s) are opened, spent fuel pool level can be maintained using the remote-manual valves from the main control room. The active components of the cooling portion of the system are powered from Class 1E sources.

The response to Question 010.021 has been revised to refer to this response.

WNP-2

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Q. 010.057
(9.1.3)

Verify that your use of the phrase "... controlled and supported to Seismic Category I requirements" means that it meets all requirements for Seismic Category I qualification.

Respone:

The cooling portion of the fuel pool cooling and cleanup system, including valves, piping, and components, meets all Seismic Category I requirements. See revised 9.1.3. Non-Seismic Category I piping systems in the reactor building are nevertheless supported to the same Seismic Category I requirements. (See Notes 10 and 32 of Table 3.2-1.)

Q. 010.058
(9.2.2)

Since the non-safety-related reactor building component cooling water system provides cooling for the reactor recirculation pumps, state the length of time that the pumps can be left without component cooling water flow before significant seal damage can occur, with consequent potential primary coolant leakage:

a. if pumps are kept running; and

b. if pumps are turned off.

Response:

Recirculation pump seal cooling is provided by both closed cooling water to the pump seal heat exchanger and control rod drive seal purge flow. If an event occurs where both pump seal cooling sources are lost, the pump seals will heat up, causing pump seal deterioration when temperatures exceed 250°F. Vendor test data, taken while operating at approximately 530°F and 1040 psia, indicate that the seals will reach 250°F approximately 7 minutes after a total loss of cooling. This will occur whether or not the pump is running.

Similar test data indicate that if one of the two seal cooling sources is operating, the pump seal temperatures will remain below 250°F and no seal deterioration should occur.

If both pump seal cooling sources fail, resulting in extreme degradation of the pump seals, the primary coolant loss has been analyzed to be less than 70 gpm. Refer to NEDO-24083, "Recirculation Pump Shaft Seal Leakage Analysis," November 1978 (Licensing Topical Report). This small amount of primary coolant leakage will be compensated for by normal or emergency water level controls. Consequently, no hazard to the health and safety of the public will result from total loss of recirculation pump seal cooling.

The position discussed above has been presented to the NRC in FSAR Appendix B, response to NUREG-0737, Item II.K.3.25.

Q. 010.059
(9.2.5)

Regulatory Guide 1.27 requires that there be sufficient water in the spray ponds for 30 days of cooling without makeup. Discuss how you will monitor the buildup of sediment on the floor of the ponds so as to assure availability of the 30-day water supply. Describe how you will clean the spray ponds without losing redundancy or degradation of the system.

Response:

Sediment buildup on the floor of the spray ponds will be monitored once every 3 months from fuel load to the first refueling outage, when the frequency of monitoring will be adjusted based on operating experience. Sediment depth will be limited to an average of 0.5 feet based on the assumption made in 9.2.5.3a for spray pond thermal analysis.

Sediment will be removed by sludge pumps utilizing hand held suction lines. Makeup water will be supplied by normal pond makeup. This method will allow cleaning of the spray ponds without losing redundancy or degradation of the system.

Q. 010.060
(9.2.6)

The FSAR states there is "... a suction head of at least 20 feet during RCIC operation" from the condensate storage tank at elevation 443′-0" and the RCIC impeller elevation 427′-3". Discuss how the 15′-9" elevational difference between the condensate storage tank and the RCIC impeller satisfies the 20′ requirement.

Response:

The 15′-9" elevation difference stated in the question is the distance between the bottom of the condensate storage tank (CST) (el. 443′-0") and the RCIC pump impeller (el. 427′-3"). The RCIC pump does not take suction from the bottom of the CST, but rather from the side of the tank at elevation 445′-4" (centerline of the suction pipe). The RCIC suction automatically transfers to the suppression pool when the water level in CST reaches an elevation of 448′ 3" (low-low level). Therefore, the minimum static head when the RCIC pump is taking suction from the CST is at least 21 feet (el. 448′ 3" minus 427′-3").

Section 9.2.6.5 states that at least 21 feet of "suction" head is available between the low-low setting and the RCIC pump impeller. Although this statement is correct, it has been revised to say "static head" instead of "suction head" to avoid confusion. The suction head available to the RCIC pump when the suction transfers at low-low level in the CST is calculated using the equation for net positive suction head (NPSH) as shown in our response to Question 022.038. Assuming a water temperature of 100°F, there is about 48 feet of NPSH available at the centerline of the pump suction which more than satisfies the 19 feet of NPSH required by the RCIC pump.

Q. 010.061
(9.3.1)
RSP

The nitrogen bottles with their associated equipment and containment instrument air system shall be a minimum of Quality Group C.

Response:

All components of the containment instrument air system from and including the outboard containment isolation valves to the main steam isolation, safety/relief, and automatic depressurization (ADS) valve operators in containment are Quality Group B. The portion of the system associated with the backup supply of nitrogen to the ADS valve operators is Quality Group C from the outboard containment isolation valves to and including the solenoid pilot valves (CIA-SPV-1A through 15A and -1B through -19B) which are mounted adjacent to the pressure reducing valves for the nitrogen bottles. See Figure 9.3-2.

The nitrogen bottles and associated control valves are standard, commercially available units that meet the requirements of Department of Transportation (DOT) and Compressed Gas Association (CGA) standards. The nitrogen bottle units are mounted per Seismic Category I, Quality Class 1 requirements. This type of nitrogen bottle assembly has proven highly reliable based on years of reactor operating experience.

The remaining components of the containment instrument air system, which are Quality Group D, are not essential for safe operation of the plant.

Q. 010.062
(9.4.1)

In your response to Question 010.029 there seems to be a contradiction between the thickness of the air intake roof slab and the height of the top of the roof slab above grade. Please clarify your numbers and provide physical drawing(s) of the air handling system with details of the remote air intake structures.

Response:

The remote air intake structure is a buried structure. Only a portion (15") of the 24-inch thick roof slab projects above grade. Please refer to sections 4298 and 4299 of Figure 3.5-52.

Q. 010.063
(9.4.1)
RSP

Discuss the control room environment which will result from the most extreme ambient and accident conditions (including the worst single failure for the HVAC). Note: The temperature/humidity for all operating/accident conditions shall be maintained with the comfort zone as defined by ASHRAE. This requirement applied to all areas which require operating personnel.

Response:

To maintain the control room at an ambient condition which is compatible to the comfort zone defined in ASHRAE, redundant seismic and environmentally qualified liquid chillers were incorporated into the control room HVAC design. Sections 9.4.1 and 6.4 have been updated. The control room is the only area with essential equipment where personnel are routinely required during accidents.

Q. 010.064
(9.4.10)

Discuss the effects of a potential failure of the non-Seismic Category I heaters in the standby service water pumphouse under the most adverse environmental conditions on the operability of the pumps.

Response:

The standby service water pumphouse electric unit heaters are designed to maintain the building above freezing during extreme environmental conditions. Failure of the unit heaters during a seismic event would not degrade the operability of the pumps. If the pumps are required to operate following a seismic event, the heat generated from the pump motor is sufficient to maintain the building above freezing. Annunciation is provided in the main control room when temperature in the pump room drops to 35°F. Appropriate action can be taken from the control room such as starting of the standby service water pump to prevent pipe freezing.

Q. 010.065
(10.4.5)

Your responses to Question 010.034 regarding the potential flooding of safety-related equipment due to a circulating water failure are inadequate. An analysis shall be conducted in accordance with Standard Review Plan 10.4.5, "Circulating Water System," which assumes:

a. An expansion joint break (note: an incident of this type occurred at an operating BWR).

b. No credit shall be taken for isolation valve closure unless these valves are designed to safety grade requirements.

Response:

Please see revised 10.4.5.3.

Q. 010.066
(4.6)

NUREG-0803 states that pipe breaks in the control rod drive hydraulic system and the resulting environmental effects should be verified on a plant specific basis. In order to conform to the guidelines of NUREG-0803, provide information to address the following concerns:

I. Taking no credit for seals and assuming no operator actions inside of the control room for 20 minutes (30 minutes for the first operator action outside of the control room plus five minutes for each additional action), provide the following information for a non-isolable break in the CRD piping between the containment penetration and the first isolation valve.

Reactor Coolant	- Mass flow rate out of the break as a function of time (= f (t))
	- Temperature = f (t)
Compartment	- Temperature = f (t)
	- Pressure = f (t)
	- Humidity = f (t)
	- Airborne Radioactivity Level = F (t)

Provide the assumptions used in determining the above information. If a computer was used, provide the computer printout and the following information:

1. With respect to the pipe to be broken:

    a. Type of fluid (water or steam);

    b. Temperature;

    c. Pressure;

    d. Source of the fluid;

    e. Flow rate (or assumed flow rate);

    f. Pipe internal diameter;

    g. Wetted perimeter of the break (feet);

    h. Total pipe internal volume;

    i. Exit flow area, if the break was not in the pipe, just described above;

j. Area of flow restriction, if any;

k. Differential elevation from the source to the pipe break;

l. Total flow resistance (only if the fluid is water);

m. Means to stop fluid flow (none, gate valve, globe valve, etc.); and

n. If 1.m is a valve, then the valve's open throat area, full open flow coefficient, valve closure time, and delay time until initiation of valve closure.

2. With respect to the compartments being analyzed:

a. Number of compartment analyzed; and

b. For each compartment:

i. initial temperature;
ii. initial pressure;
iii. initial humidity;
iv. free air volume (cubic feet);
v. number of vents and vent areas (square feet) for each vent; and
vi. minimum pressure to initiate flow to the next compartment (psia).

3. All assumptions used, including but not limited to the:

a. Orifice coefficient for the "end effects" for the discharge fluid; and

b. Fluid expansion factor.

II. Verify that all electrical and mechanical equipment needed to mitigate the event is qualified to the environmental conditions determined in Part I. Verify that no pump cavitation will occur when pumping 212°F water.

III. Provide a discussion of the procedural steps to be taken to isolate the break in the CRD pipe at the outside surface penetration of containment to terminate the small pipe LOCA accident. Identify all equipment and materials required to isolate the break. Provide a commitment to maintain onsite these items as dedicated equipment and

materials. Discuss your procedure to verify periodically the existence and condition of these dedicated items.

IV. Assuming that the sumps are inoperative for the event in Part I (since they are not Seismic Category I, Class 1E), provide maximum water level in the compartment. Verify that no equipment required to bring the plant to a safe condition will fail as the result of internal flooding. Verify that no personnel radiation hazard will exist due to wading through reactor coolant.

V. Verify that all analysis performed in Parts I and IV includes time for the items listed below. The first action outside of the control room should be assumed to be at least 30 minutes after annunciation in the control room plus five minutes for each personnel action in accordance with ANS 58.8. Installing scaffolding requires multiple actions just like donning protective garb.

1. HP survey of the area and documentation;

2. Establishment of protective garb requirements;

3. Establishment of change areas, clean areas, check-in and check-out lists, waste disposal and facilities and transportation of necessary garb to the area for the workers;

4. Following all HP procedures;

5. Review of repair procedures;

Assume that the event occurs with the minimum plant personnel available on any shift.

Response:

Following the Supply System submittal responding to NUREG-0803 concerns (Reference 1 and 2), the NRC is requesting, via NRC Question 010.066 (4.6), that the Supply System evaluate a single CRD withdrawal line break at a specified location. However, according to the NRC's own NUREG-0803:

> From page 2-2 . . . This assessment is based on the fact that the CRD withdraw lines penetrating the containment and routed to the HCUs are small in diameter (3/4 in.) and are conservatively designed and of high quality. Nevertheless, even if the staff postulated a break in one of these lines during reactor operation (including scram):

(1) The leakage through this break is within the reactor coolant makeup capabilities (feedwater and reactor core isolation cooling) since, as required by GDC 14, "Reactor Coolant Pressure Boundary," the CRD system contains redundant CRD seals and a restricted flow area that limits the reactor coolant leakage to a very small value;

(2) The reactor can be shut down and cooled down in an orderly manner; and

(3) No leakage from the SDV, where flow from all other CRD withdraw lines is accumulated following scram, will occur through the break because of the existence of a check valve between the SDV and the withdraw line manual isolation valves.

and from page 4-27 . . . Breaks upstream from the isolation valves in the 3/4-inch piping were judged to be minor in size and with no potential short-term effects on the core cooling capability.

The Supply System agrees that a single CRD withdrawal line rupture event is minor and fully enveloped by existing analyses. For example, an instrument line rupture, 15.6.2 and 6.3.3, presents a similar type break, but with larger leak flow and more severe consequences. However, as requested in NRC Question 010.066 (4.6), evaluation of the single CRD withdrawal line rupture is presented in the following paragraphs to provide the NRC reviewers with quantification of the event to support the already recognized conclusions stated above.

Question 010.066 contains five parts (I through V) and each section or part is addressed separately below:

I. A simplified diagram of the postulated line rupture is shown in Figure 010.066-1. It is assumed that a complete severence of a withdrawal line occurs, following a reactor scram, at a location between the reactor containment and the first isolation valve (in HCU). There is no back-flow through the break due to a check valve between the break location and the scram discharge volume (SDV). Therefore, the only leak out of the break is the reactor coolant that can leak around the CRD piston seals. This leakage is normally about 2 gpm and can be as high as 6 gpm with badly worn seals. The stipulation in the Question 010.066 above that no credit for seals be taken is an unrealistic conservatism. However, GE tests show that with seals completely missing, at operating pressure,

the leakage is a maximum of 10 gpm. This value is used as the initial leak rate.

The scenario can be defined as follows:

Time	Event Description
0	Reactor scrams. Complete severence of a single CRD withdrawal line occurs.
20 mins.	Operator initiates cooldown at 100°F/hr due to leakage exceeding tech spec values.
240 mins. (3.8 hr.)	Reactor on RHR with RC temp $\leq$ 200°F

The analysis assumptions and results are presented below in the same order and numbering system used in the question.

1. a. Type of fluid: Water

   b. Temperature: } See Figure 010.066-2

   c. Pressure: }

   d. Source of fluid: Reactor vessel

   e. Flow rate: See Figure 010.066-3

   f. Pipe internal diameter: }

   g. Wetted perimeter: }

   h. Total pipe internal volume: } Not applicable, see Item 1.j below.

   i. Exit flow area: }

   j. Area of flow restriction: 1.19×10^{-4} ft^2 - equivalent area for 10 gpm leak at 1000 psia.

   k. Differential elevation: See Figure 010.066-1

   l. Total flow resistance: Total pipe resistance was neglected for the low flow rates encountered. The hydrostatic analysis accounted for orifice pressure drop at the CRD, see Item 3.a below.

m. Means to stop fluid flow: Mechanical plugging at operator's discretion, see Part III below.

n. Not applicable.

2. With respect to the compartments being analyzed:

a.} b.} Compartment pressure and temperature were not analyzed for two reasons. First, as stated in our original response to NUREG-0803, no equipment necessary to achieve cold shutdown following a scram is located at the 522' level; therefore, compartment conditions for equipment qualification is not a concern. Secondly, the integrated leak mass at 3.8 hours is only 9370 lbm which, even if held to the 522' compartment, would not significantly affect pressure or temperature. After 3.8 hours, any leakage (< 2 gpm) is at a temperature of less than 200°F and has negligible effect.

3. All assumptions used, including but not limited to the:

a. Orifice coefficient: Moody critical flow was assumed out to 3.8 hours for an area equivalent to no seals and 10 gpm initial flow. For the hydrostatic pressure difference analysis, an orifice equation was used with a discharge coefficient of 0.7.

b. Fluid expansion factor: Not applicable.

II. As stated in Part I.2 above, no equipment necessary for safe shutdown following scram is located in this area, therefore, environmental qualification is not an issue. The Supply System interprets ". . . no pump cavitation . . ." to refer to the sump pump to which the reactor leakage drains (see Figure 010.066-1). If the total leakage (assuming no flashing) drains to the sump, its water level would be 0.2". Therefore, pump cavitation is not an issue as the sump pumps are not required to prevent flooding.

III. After 3.8 hours the leak is reduced to $\leq$ 1.7 gpm and 200°F water from the severed withdrawal line. It should be noted that a leak of 1.7 gpm is possible only by neglecting all withdrawal line losses and assuming no CRD piston seals exist. The realistic flow rate would be much less. A temporary fix could be effected easily by plugging or freeze sealing the tube. Since a permanent fix becomes a question of availability and the total scenario given in Part I indicates no safety questions pertain to the incident, the Supply System does not feel a commitment to maintain and verify the existence of the equipment necessary to fix a leak of less than 1.7 gpm is prudent or necessary.

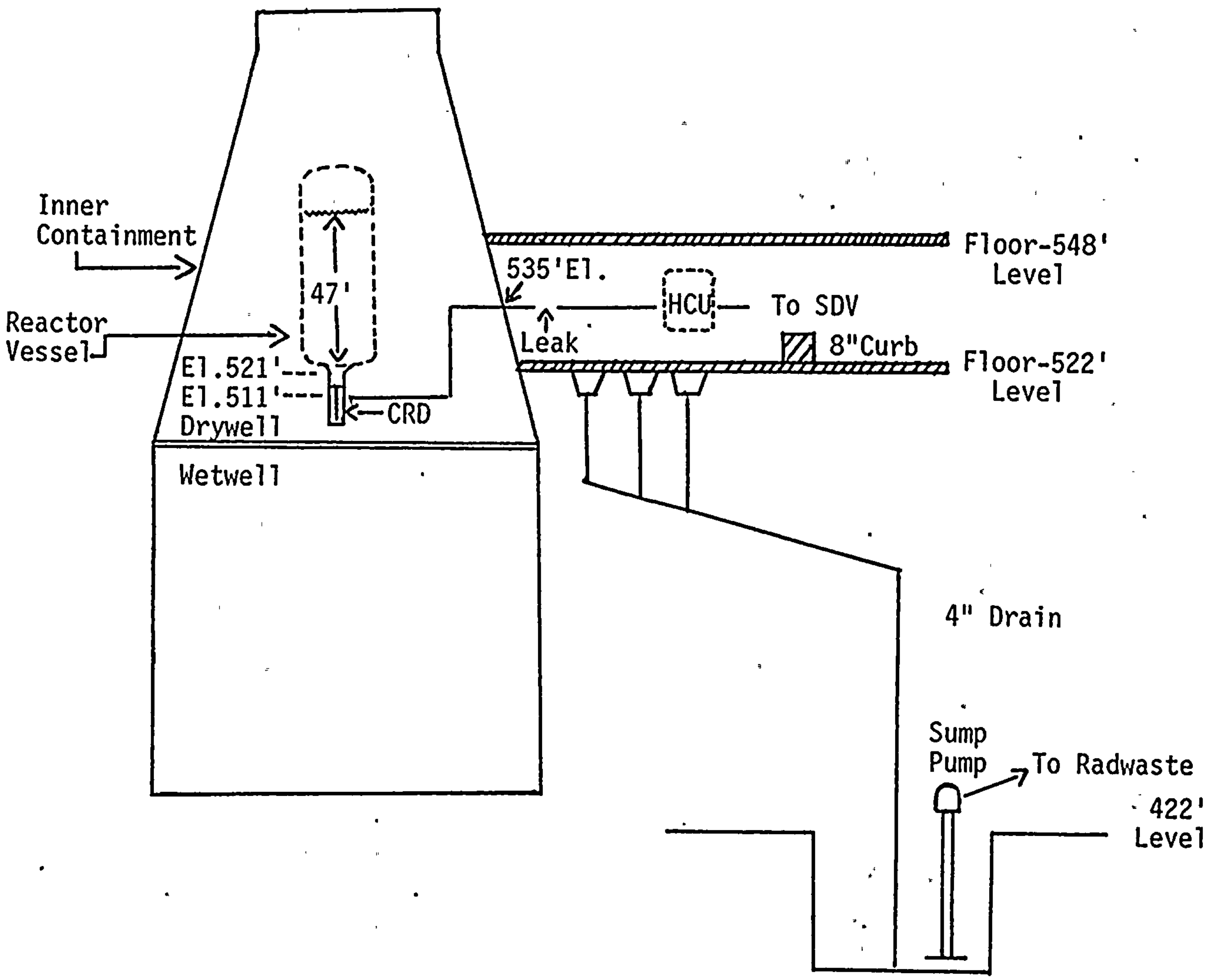
IV. See response to Part II of this question regarding water level and equipment necessary to achieve a cold shutdown. With the leak at $<$ 2 gpm and floor drains available, no personnel will be required to 'wade' through reactor coolant. As was true for Part III of this question, it is a matter of plant availability, not public health and safety, as to when personnel would enter. Access to the area can be controlled since no time constraints are imposed for stopping a leak of this magnitude.

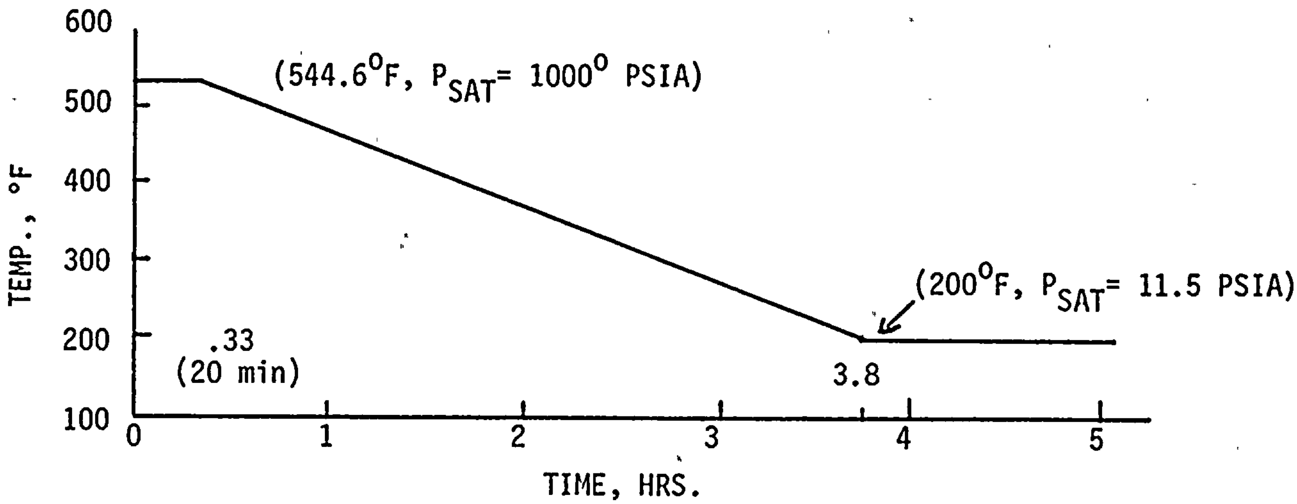
V. Parts III, IV, and V of this question carry the connotation that this scenario should be reacted to on the old event-based procedure basis. WNP-2 has formulated its secondary containment response activities in accordance with the BWR Owner Group symptom based procedures which adequately cover coolant leakage events including small leaks such as postulated here. As mentioned in Parts III and IV of this question, it would not be meaningful to assign times to each action as the mitigation function can be done according to availability determination and is not a question of necessity for public health and safety.

Additional Information

During a telephone conversation January 13, 1983 between the NRC and the Supply System, the capability of the curbed area around the control rod drive line piping to contain water from CRD line break was questioned. The following information was provided.

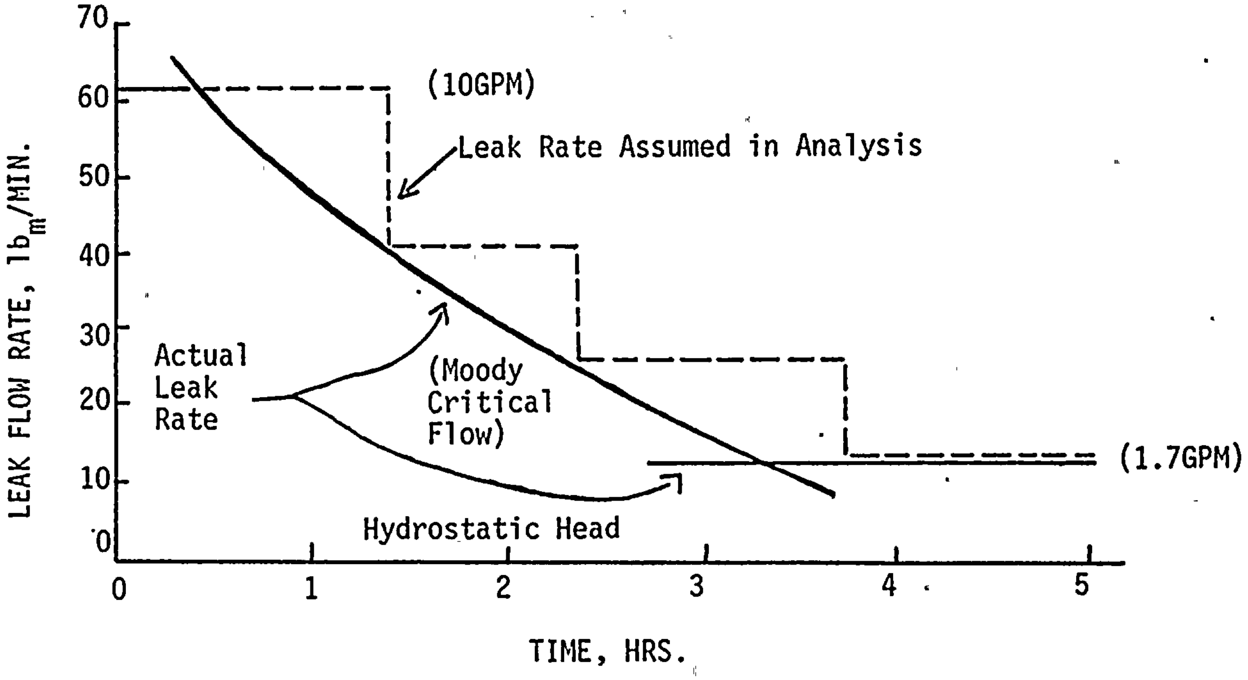
At the time cooldown and depressurization is complete (3.8 hours), a total of 9370 lb_m of coolant has been released from the break (see I.2.a and I.2.b of the response to this question). This is equivalent to 160 ft^3 of water. The volume available within the curbed area is about 700 ft^3. Therefore, the coolant is retained within the curbing.


Inner
Containment
Reactor
Vessel
47'
El.521'
El.511'
Drywell
CRD
Wetwell
535'El.
Leak
HCU
To SDV
8"Curb
Floor-548'
Level
Floor-522'
Level
4" Drain
Sump
Pump
To Radwaste
422'
Level



TEMPERATURE VERSUS TIME WITH
100° F/HR COOLDOWN

FIGURE
010.
066-2



LEAK RATE VERSUS TIME

FIGURE 010.066-3

Q. 010.067
(3.5.1)

The FSAR infers that compartment walls will stop internally generated missiles. Verify that all compartment walls will: (1) prevent missiles from penetrating the walls, and (2) not form any secondary missiles, either by spalling or scabbing.

Response:

All credible missiles, as listed in FSAR Table 3.5-7, were analyzed to determine if any credible missile could impact a concrete wall and penetrate the wall or form secondary missiles by spalling or scabbing.

This analysis was carried out using the methods outlined in Bechtel Topical Report BC-TOP-9A, "Design of Structures For Missile Impact", Section 2.1.2, dated September 1974.

The list of credible missiles was divided into three types of missile sources, small centrifical fans, large variable pitch fans, and the RCIC Turbine drive. The potential missile energy level from each missile source was compared and bounding, representative missiles were analyzed in detail. The concrete thickness required to preclude scabbing or spalling were calculated and compared with the scheduled wall thicknesses in all areas with credible missiles. For all representative missiles, the concrete thickness required to preclude spalling and scabbing is less than one foot by a substantial margin. The thinnest concrete wall in a credible missile area is one foot thick. This supports the concrete wall integrity inferred in the FSAR.

Q. 010.068
(3.5.1)
(RSP)

The FSAR states that "secondary missiles are not considered credible due to their low probability of occurrence and their low kinetic energy levels. In addition, no reliable method to predict secondary missile characteristics is known." Regarding your statement that no reliable method to predict secondary missile characteristics, the staff requests you use the empirical formula generated by CEA-EDF and Bechtel Topical Report BC-TOP-9A as appropriate to calculate the spalling and scabbing with the missile striking perpendicular to the barrier surface. Provide example calculations using each method. For each occurrence of scabbing, verify that all redundant equipment is completely protected from debris of all sizes.

Response:

We have reviewed both papers referenced in your question. The paper you refer to as "CEA-EDF" is, to the best of our knowledge, available only in French. If it should be determined that this paper is available in English, we will review it after staff approval and determine if it has any applicability to our project.

Bechtel Topical Report BC-TOP-9A does not provide any information on secondary missile characteristics. Formulas are presented to determine if secondary missiles would occur due to primary missiles striking concrete walls. However, an evaluation of a secondary missile would require a method of determining the size, trajectory, and velocity of the secondary missile.

Example calculations of how to predict spalling and scabbing are contained in BC-TOP-9A, which has been approved by the staff. The method outlined in BC-TOP-9A was used to evaluate all credible missile impacts. Except for the RCIC turbine, there are no credible missiles with sufficient energy to cause spalling or scabbing, thereby negating the necessity of evaluating the characteristics of the secondary missiles. See response to NRC Question 010.067 for the status of the RCIC turbine missile analysis.

Q. 010.069
(3.5.1)

With respect to bolted valve bonnets, the FSAR does not address the simultaneous failure of the bolts due to chemical attack followed by a pressure transient. Verify that no valves with bolted bonnets are located under or near any pipe or vessel containing any chemically corrosive material. For any valve for which this cannot be shown, specify the valve and provide a description of the design provisions which will be provided to ensure that no chemicals will be able to come in contact with the bolts. As an alternative, consider the bonnet to be a missile and verify that redundant safety-related equipment needed for safe reactor shutdown will not be damaged, including damage due to secondary missiles.

Response:

There are no corrosive chemicals in the plant that can come in contact with valve bonnet bolts, or any safety-related components in the plant.

Q. 010.070
(3.5.1)

The FSAR states that thermowells and detectors "are evaluated as potential missiles if a single circumferential weld would cause their ejection." Within the same paragraph the FSAR states that "because of their highly conservative design thermowells and sample probes are not considered credible missiles." These two statements are contradictory. Verify: (1) that thermowells and sample probes were evaluated as missiles, and (2) that no safety-related equipment needed for safe reactor shutdown would be damaged by thermowell or sample probe missiles including damage by secondary missiles.

Response:

Thermowells and sample probes which are retained by a single circumferencial weld were evaluated as potential missiles. A detailed analysis was made to determine both the actual stresses and yield shear and tension stresses for each thermowell and sample probe utilized. Thermowells and sample probes are retained by full penetration welds and are radiographed to insure integrity. The yield stress values were found to range from 6.6 to 20 times the actual stress values. This is a very high margin of safety, and supports our conclusion that thermowells and sample probes are not credible missiles because of their highly conservative design and weld quality inspections.

Q. 010.071
(3.5.1)
(RSP)

Compressed gas bottles are potential missiles. Consider all compressed gas bottles and accumulators as potential missiles and verify that redundant safety-related equipment needed for safe reactor shutdown will not be damaged, including damage by secondary missiles. If damage will result, describe the design provisions which will be used to protect the redundant equipment.

Response:

Compressed gas bottles and accumulators were evaluated as potential missiles. A detailed analysis was made of the gas bottles and accumulators to determine a credible method of failure. The actual yield shear and tension stresses for the gas bottle and accumulator stem connections were calculated. The yield stress values were found to be 89 times the actual stress values. All bottles and accumulators are held in place by at least two seismically qualified methods of retention. This analysis is the basis of our conclusion that compressed gas bottles and accumulators are not credible sources of missiles because of their highly conservative design.

Q. 010.072
(3.5.1)

The FSAR states that "when the separation and redundancy of the essential systems is not adequate ... It is shown that the essential components will not be damaged by the credible missile." Provide a discussion of, and figures as appropriate to illustrate, how "it is shown" that no damage would be incurred. Provide sample calculations, if appropriate.

Response:

The FSAR paragraph addressing this question is 3.5.1.1.4b(3). This information was transmitted in letter G02-82-492, from Mr. G. D. Bouchey to Mr. A. Schwencer, dated May 28, 1982.

The analytical methods used to show that essential components will not be damaged when struck by the credible missile are contained in 3.5.1.1.4c and Bechtel Topical Report BC-TOP-9A, "Design of Structures for Missile Impact", Revision 2, dated September 1974. Table 3.5-7 tabulates credible missiles and notes measures taken to ensure essential systems are not damaged.

Q. 010.073
(3.5.1)
(RSP)

Provide the results of an analysis which verifies for each rotating piece of machinery that no failures can occur due to metal fatigue or chemical attack (such as chloride stress corrosion) such that any piece can become separated and, therefore, a missile. As an alternative, verify for each piece of rotating machinery that the casing will contain any missile generated at its maximum kinetic energy or that no safety-related equipment will be damaged by the generation of the missile taking no credit for any casing or enclosure.

Response:

For each rotating piece of equipment, the rotating component was assumed to fail regardless of fatigue useage level. For each failure, an analysis was performed to determine if the casing or protective housing would contain the resulting missile, or an analysis was performed to determine that no equipment required to safely shutdown the plant could be damaged. Details of the method used and the results of the analysis of each piece of rotating equipment are listed in Table 3.5-7.

Q. 010.074
(3.5.1)

Provide sample calculations and reference any support documentation used to assure that HVAC fan casings will retain any and all internally generated missiles.

Response:

A detailed analysis was performed to determine if the HVAC fan blades could potentially penetrate the fan casing in the event of a fan blade failure or hub failure. The results of the analysis showed that fan casings retained all internal missiles, see Note B of Table 3.5-7. The calculation method used is described in detail in Bechtel Topical Report BC-TOP-9A, "Design of Structures for Missile Impact", Revision 2, dated September 1974, Section 2.2.

Q 022.1

It is our position that the design of the containment purge system should comply with the recommendations in Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operation." Accordingly, the purge system should either conform with BTP CSB 6-4 or operation of the system should be limited to less than one percent of plant operation time.

Response:

Please see 6.2.1.1.8 for the requested information.

Q. 022.002

In 6.2.1.7 of the FSAR, you state that the instrumentation required to monitor containment parameters is discussed in 7.6.1.8. However, 7.6.1.8 discussed the rod sequence control system. Correct this cross-reference.

Response:

The cross-reference has been corrected. 6.2.1.7 now refers to 7.5.1.5.

Q 022.003

Provide the maximum calculated negative pressure following the inadvertent actuation of the containment spray system. Indicate the design pressure for this condition. Provide a description of the analytical model and provide justification to demonstrate that the assumptions used to determine the internal containment pressure response are conservative.

Response:

The requested information is in 6.2.1.1.4.

Q 022.004

Identify the types of insulation used within the containment (e.g., reflective metal insulation, mass insulation, and encapsulated or sheathed mass insulation). Discuss the extent to which insulation in the vicinity of the postulated pipe break could be stripped from piping and components and identify the type of insulation involved. Discuss the potential for loose insulation, and other debris, to clog the vents leading to the suppression pool and to the suction screens of the emergency core cooling systems.

Response:

Please see 6.2.1.1.2 for the requested discussion.

Q. 022.005

In the event of a pipe rupture inside a major component subcompartment (e.g., the annulus and the upper head region), the initial blowdown transient would lead to nonuniform pressure loadings on both the structure and the enclosed component(s). Perform a multi-node, pressure response analysis of the subcompartments cited above as well as any other major component subcompartment. Provide the following information with respect to these analyses:

a. Indicate the type, area, and location of the pipe break for each analysis, including the basis for each parameter. Specify whether the pipe break was postulated for the evaluation of the subcompartment structural design, component support design or both.

b. For each subcompartment, provide a table of the blowdown mass flow rate and the energy release rate as a function of time for the break which results in the maximum structural load, and for the break which was used for the component support evaluation.

c. Provide a schematic drawing showing the compartment nodalization for the determination of maximum structural loads, and for the component support evaluation. Provide sufficiently detailed plan and section drawings for several views, including the principal dimensions, showing the arrangement of the compartment structure, major components, piping, and other major obstructions and vent areas so as to permit verification by the NRC staff of the subcompartment nodalization and vent locations.

d. Provide a tabulation of the nodal net volumes and interconnecting flow path areas. For each path, provide an L/A (ft^{-1}) ratio, where L is the average fluid flow distance in that flow path and A is the effective cross-sectional area. Provide values of vent loss coefficients and/or friction factors used to calculate the flow between nodal volumes, including justification for the selected values. When a loss coefficient consists of more than one component, identify each component, its value and the flow area to which the loss coefficient applies.

e. Describe the nodalization sensitivity study which you performed to determine the minimum number of nodal volumes required to conservatively predict the maximum pressure load acting on the subcompartment structure. This nodalization sensitivity study should include consideration of spatial pressure variation (e.g., pressure variation circumferentially, axially and radially within the compartment). Describe and justify the nodalization sensitivity study performed for the evaluation of the major component supports where transient forces and moments acting on the components are of concern.

f. Discuss the manner in which movable obstructions to vent flow (e.g., insulation, ducting, plugs, and seals) were treated. Provide analytical and experimental justification for the assumption that vent areas will not be partially or completely plugged by displaced objects. Discuss how the dimensions of the insulation for piping and components were considered in determining volumes and vent areas.

g. Graphically show the pressure (psia) and differential pressure (psi) responses as functions of time for each node. Discuss the basis for establishing the differential pressures on structures and components.

h. For the subcompartment structural design pressure evaluation, provide the peak calculated differential pressure and the time of peak pressure for each node. Discuss whether the design differential pressure is uniformly applied to the compartment structure or whether it is spatially varied.

If the design differential pressure varies depending on the proximity of the pipe break location, discuss how the vent areas and flow coefficients were determined to provide assurance that regions remote from the break location are conservatively designed.

i. Provide the peak and transient loading on the major components used to establish the adequacy of the support design, including the load forcing function and transient moments (e.g., $F_x(t)$, $F_y(t)$, $F_z(t)$, $M_x(t)$, $M_y(t)$, and $M_z(t)$) resolved about a specific, identified coordinate system. Provide the projected area of the major components used to calculate these loads and identify the location of that area projection on plan and section drawings in the selected coordinate system. This information should be presented in such a manner that confirmatory evaluations of the loads and moments can be made by the NRC staff.

Response:

See revised FSAR 6.2.1.2.

Q. 022.006

For bypass leakage of the secondary containment provide the following:

a. Revise Table 6.2-16 to include the numerical value of the leakage calculated for penetrations where the potential for such leakage exists.

b. Describe the analytical method by which this bypass leakage was calculated.

c. Indicate if there are any guard pipes which pass through the secondary containment. If so, provide justification for not including this source of potential bypass leakage in Table 6.2-16.

d. Where a closed system is relied on as a barrier, discuss how it meets the requirement specified for a closed system in 8.9 of BTP-CSB 6-3.

Response:

a. Table 6.2-16 has been revised.

b,c. Please refer to 6.2.3.2 and 6.2.3.3 for requested information.

d. No closed systems are relied on as barriers.

Q. 022.007

Provide the secondary containment pressure time response for the design basis accident. List and discuss all assumptions made in this analysis.

Response:

See revised 6.2.3.3 and Table 6.2-29.

Q. 022.008

Identify the manufacturer of the hydrogen recombiner and describe the test program which demonstrates that the hydrogen recombiner will perform as required in the containment environment following a postulated loss-of-coolant accident. Provide the systems quality group classification including that of the hydrogen analyzer.

Response:

The hydrogen recombiner was manufactured by Air Products and Chemicals, Inc. The tests are briefly discussed in 6.2.5.4 and detailed, supplemental information is being provided by separate letter to the NRC. The analyzer including quality group classification, is discussed in 7.5.1.5. All pressure containing equipment, including piping between components, is considered an extension of the containment and is classified code Quality Group B, as discussed in 6.2.5.2.3.

Amendment 10, revising Chapter 7, moved the discussion of the hydrogen recombiner to 7.3.1.1.8.

Q 022.009

Provide plan and evaluation drawings of the personnel air lock, and identify all mechanical and electrical penetrations. Discuss and show schematically the design provisions that will permit the personnel air-lock door seals and the entire air-lock to be tested. Discuss the capability of the door seals to be leak tested at a pressure of Pa (i.e., the calculated peak containment internal pressure). If it will be necessary to exert a force on the doors to prevent them from becoming unseated during leak testing, describe the provisions for doing this and discuss whether or not the mechanism can be operated from within the air lock.

Response:

Please see 3.8.2.7.5 for information requested.

Q. 022.010
(6.2.6)

DELETED

Q . 022.011

For each piping penetration that does not conform to the explicit requirements of the General Design Criteria of 10 CFR Part 50, provide a description of your proposed design basis. Augment the information provided in the FSAR to include the following:

a. Justification for locating both isolation valves outside containment;

b. An acceptable second isolation barrier where a single valve is indicated; and

c. Describe the information provided to the plant operator, which will enable him to decide if he will need to operate a remote manual valve in the event of a loss-of-coolant accident.

Response:

a. Please see Table 6.2-16.

b. Please see Table 6.2-16.

c. Please see 6.2.4.3.4.

Q 022.012

For those isolation systems which satisfy the requirements of Criterion 57 of the Design Criteria, provide adequate information to demonstrate that these systems meet the requirements of a closed system as stated in Section 8.9 of BTP-CSB 6-3.

Response

There are no lines presently listed as falling under GDC 57.

Q 022.013
(6.2.1)

Provide the following additional information with respect to the secondary containment:

a. Appropriate plant elevation and section drawings for those structures and areas that will be maintained at negative pressure following a postulated loss-of-coolant accident and that were considered in the dose calculation model.

b. Your proposed technical specification limit for leakage which may bypass the filters of the standby gas treatment system (e.g., valve leakage and guard pipe leakage); and

c. A discussion of the testing methods which will be used to verify that the systems provided are capable of reducing and maintaining the pressure within all secondary containment volumes to a negative pressure of 0.25 inches (water gauge).

Response:

Information on the secondary containment is provided in 6.2.3; secondary containment functional design. Specifically,

a. The reactor building is the secondary containment, refer to Figures 1.2-3 thru 1.2-7 and 3.8-1 and -2.

b. Refer to response to NRC question 22.6. The isolation valves on the lines which have been identified as potential secondary containment bypass paths will be tested to ensure the total bypass leakage rate is equal to or less than the amount calculated in 6.2.3.3.

c. Refer to 6.2.3.4 and 7.3.1.1.7. The reactor building contains sufficient openings to maintain uniform pressure throughout the building, refer to previously listed figures.

Q. 022.014

Note 2 of Table 6.2-16 states that the suppression pool serves as an isolation barrier to the environment. It is our position that this is unacceptable.

Response:

The penetrations that reference note number 2 in Table 6.2-16 meet General Design Criteria 56 and include a single containment isolation valve and are closed systems outside containment. This provides double barrier containment isolation and the water in the suppression chamber is not considered as part of containment isolation. The closed systems outside containment are protected from missiles, are Seismic Category 1, Safety Class 2, and have a design temperature and pressure at least equal to that of the containment.

Table 6.2-16 has been revised to include the General Design Criteria for all penetrations. (See response to Question 022.027).

Q 22.15

Provide a detailed drawing showing the locations of the containment spray headers relative to the internal structures.

Response:

The containment spray system is discussed in 6.5.2. The suppression chamber spray header is shown in Figure 3.5-3. The lower drywell spray header is shown in Figures 3.5-4, 3.5-21, 3.5-22, 3.5-25, 3.5-26, 3.5-27 and 3.5-28. The upper drywell spray header is shown in Figures 3.5.16, 3.5-17, and 3.5-18.

Q 22.016

Provide the value of the external design pressure for the containment structure.

Response:

The primary containment is designed for a total external pressure of 4 psid; however, since the compressed insulation between the concrete biological shield and the containment exerts a uniform 2 psid external pressure - half of the total external pressure differential allowed - the reactor building pressure may be no greater than 2 psi above the primary containment pressure. This value is given in Table 6.2-1, "Containment Design Parameters", and 6.2.1.1.2.

Q 022.017

Provide the assumptions and initial conditions for the activation of both drywell spray loops following a postulated loss-of-coolant accident (LOCA) that has purged all the drywell noncondensable gases into the suppression chamber; provide the same information for the inadvertent drywell spray activation at normal operation conditions. This information is requested so that we may perform an independent evaluation of the reverse differential pressure across the drywell floor to establish the degree of conservatism in the small steam line break analysis.

Response:

Inadvertent actuation of drywell sprays is not considered a credible event at WNP-2 since it requires more than one single failure or operator error. FSAR Section 6.2.1.1.4 discusses the analysis and assumptions used for containment negative pressure evaluation. Initial conditions are given in Table 6.2-19, Assumptions and Initial Conditions Employed in Negative Pressure Design Evaluation, and Table 6.2-1, Containment Design Parameters.

Q 022.018
(6.2.1)

We require your compliance with our proposed position on containment steam bypass for small breaks. The details are contained in the Containment Systems Branch Technical Position, "Steam Bypass for Mark II Containments," a copy of which is enclosed.

Response:

Please see response to question 31.070.

Q 22.019
(6.2.1)

Provide the information requested in paragraphs B.1.a, B.1.d, B.1.g, B.4 and B.5 of Branch Technical Position CSB 6-4, "Containment Purging During Normal Plant Operation".

Response:

Refer to the response to previous question 22.1. The valves in the containment purge and vent lines are containment isolation valves. The design criteria for the isolation valves is discussed in 6.2.4, Containment Isolation System, and 6.2.6, Containment Leakage Testing. In addition, paragraph 6.2.1.1.8.2 has been revised to expand the definition of reactor operation and include information relative to purge isolation valve qualification.

Q 22.020
(6.2.1)

Provide the test results and method of analysis utilized to determine the seismic sloshing loads discussed in 6.2.1.1.3.1 of the FSAR. Provide the basis for the acceptance of these loads.

Response

Please refer to 6.2.1.1.3.1, 3.8.2.4.3 and 3.8.2.4.3.2 for the information requested.

Q 22.021
(6.2.1)

With regard to all safety-related equipment located inside the containment building such as the control rod drive hydraulic system, the reactor vessel supports and all incore instrumentation leads, we require that the environment be maintained within the maximum temperatures and rate of temperature changes for which the equipment is qualified to operate. Indicate whether the reactor building ventilation system (RBVS) is required to assist in maintaining an acceptable temperature range. If it is, provide the following information regarding the RBVS:

a. Justification for not treating this system as an engineered safety feature.

b. The results of an analysis that the RBVS will not be a potential source of missiles; demonstrate that the RBVS meets our pipe whip criteria.

c. A discussion of the operating procedures to be initiated in the event that the RBVS is unavailable.

d. The location of all temperature sensors associated with the operation of the RBVS.

Response:

As discussed in 6.2.3 and 9.4.2 the RBVS is the ventilation system for the secondary containment and is designed to be automatically shutdown and isolated on receipt of an accident signal. Except for the isolation portion of the RBVS, the system is not an engineered safety feature. All equipment in the reactor building required to bring the reactor to a safe shutdown is either qualified to the accident environment of the secondary containment or enclosed in compartments which are cooled by the reactor building emergency cooling systems (see 3.11.1.2, 3.11.4.2, and 9.4.9). Seismic Class I supports are provided for all RBVS and emergency cooling system ducting to preclude damage to any safety related components in the secondary containment during a seismic event.

The primary containment is cooled by the primary containment cooling system as discussed in 9.4.11. It should be noted that this system is not required for safe shutdown and is not assumed to be operable for accident analysis or assessment of the primary containment accident environment.

Q. 022.022

Provide a detailed justification for the use of Quality Group C valves for containment isolation. Our position on this matter is that Quality Group B valves should be used.

Response:

All containment isolation valves are either Quality Group A or B, as shown on Table 6.2-16. (Reference response to question 022.027).

Q 22.023
(6.2.5)

Identify the seismic category and the quality group of the hydrogen monitoring system.

Response:

The hydrogen monitoring system is Seismic Category I. The system is Quality Group B for the suction lines to the downstream side of the excess flow check valves and for the discharge line from the upstream side of the check valve. Between these code breaks, including the monitor, the Quality Group is D.

Refer to Figure 3.2-15, paragraphs 6.2.5.1 and 6.2.5.2.3, and the response to question 22.8 in Volume 21.

Q.022.024 Deleted

Q. 022.025

Identify the location of the hydrogen sampling points in the drywell and the suppression chamber. Identify the location of the suction and discharge points of the combustion gas control system with respect to local structures and equipment.

Response:

Please refer to FSAR Table 7.6-12 and Figures 6.2-32 through 6.2-35 for the requested information.

Amendment 10, revising Chapter 7, changed the number of Table 7.6-12 to 7.5-2.

Q. 022.026

DELETED

Q. 022.027
(6.2.6)

Augment Table 6.2-16 to provide the information requested in Section 6.2.4.2, "Systems Design" of Regulatory Guide 1.70.

Response:

Tables 6.2-13, 6.2-16 and 7.3-13 have been combined into one table, number 6.2-16. This table has been further expanded to include all the information required in Section 6.2.4.2 of Regulatory Guide 1.70. Chapter 6 will be appropriately revised to reflect the revised Table 6.2-16.

Table 7.3-13 has subsequently been replaced by a new table not pertinent to this question. Chapter 6 has been revised to reflect the revised Table 6.2-16.

Q 22.028

Several of the loads presented in Table 3.4-1 of the Plant Design Assessment Report (DAR) have been generated using computer codes which have not been reviewed by the NRC staff. Provide a complete description of your method of analysis for all codes presented in Appendix D of the DAR.

Response

Three computer codes have been used to develop short term LOCA hydrodynamic loads for WNP-2 plant assessment. The three codes are the downcomer vent clearing analytical model computer code VENT, the pool swell analytical model computer code SWELL, and the LOCA bubble charging analytical model computer code BUBBLE. Complete documentation on each of the above three codes is provided or referenced in Appendix D of the DAR (see below for specific references). Documentation on the WNP-2 load calculation procedure is provided in 3.2.1 of the DAR.

1. Vent Code

   a) Assumptions

      See Section D.2 of the DAR.

   b) Equations

      See Figure D-1 of the DAR.

   c) Methodology

      See Figure D-1 of the DAR.

2. Swell Code

   The assumptions, equations, and methodology for the Swell Code are identical to that described in Reference D-1.

3. Bubble Code

   The assumptions, equations, and methodology for the Bubble Code are identical to that described in Reference D-7.

Q. 022.029

Safety issues such as the proposed reductions in the pool boundary chugging loads and the safety/relief valve quenchers loads are being resolved generically and are not scheduled for resolution until about 1980. Discuss your short term solutions for these types of problems which require a long period of time to be resolved.

Response:

For final resolution of chugging and safety/relief valve discharge loading issue, refer to the Plant Design Assessment Report (DAR Revision 3) in Appendix G to this FSAR.

Q 22.030

Provide a more detailed description of the significant modifications to the WNP-2 facility which are being made or scheduled to be made based upon the results of your ongoing experimental and analytical efforts.

Response

Modifications made to the WNP-2 plant as a result of investigations of the SRV and LOCA phenomenon are listed below:

1) Seven horizontal ring tee stiffeners added to the submerged circumference of the steel containment vessel.

2) Redesign of the downcomer bracing system from a system of radial beams to a pipe truss system which includes braces for the suppression pool columns and lateral restraints for the SRV piping.

3) Revised location of platforms in the suppression pool and revision of their connection to the containment vessel.

4) Revised location of wetwell to drywell vacuum breakers.

5) Provided a quencher discharge device and support tower for each SRV line. Added a redundant vacuum breaker on each SRV discharge piping line. SRV piping was rerouted to optimize the line air volumes.

6) Provided additional stiffening in the area of many containment vessel piping penetrations.

7) Redesign of piping systems resulting in additional pipe supports and snubbers.

8) Downcomer pipes were locally reinforced where the SRV pipe penetrates the downcomer wall and the downcomer flanges were removed.

9) Installation of a suppression pool temperature monitoring system.

10) Revised the design of piping suction strainers.

Q. 022.031

Provide a detailed calculation of the friction loss coefficient for the entire vent system. Indicate whether the results of the 4T (temporary tall test tank facility) portion of the ongoing generic Mark II test program have been used to confirm the calculation vent loss coefficient. Additionally, indicate the margin applied to the calculated friction loss coefficient to account for any differences between the WNP-2 vent design and that of the 4T test facility.

RESPONSE:

The WNP-2 downcomer loss coefficient has been recalculated, using data from References 1, 2, and 3, to be 2.59.

The downcomers extend above the drywell floor 1.5 inches. Each downcomer has a deflector plate 10.75 inches above the downcomer supported by four one-inch thick stiffener plates that are six inches apart center to center. Figure 6.2-24 of the FSAR (Reference 1) shows the inlet geometry of a typical 24-inch downcomer. Every downcomer has a total length of 45 feet 3.25 inches with a wall thickness of 0.375 inch. There are 84 24-inch diameter downcomers and 18 28-inch downcomers. A 10-inch SRV discharge line is located concentrically through each 28-inch downcomer and extends up through the deflector plate and exits out the side of the 28-inch downcomer eight feet two inches below the top of the downcomer.

The total loss coefficient for the 24-inch and 28-inch downcomers has been broken down into three and four components respectively as follows:

Inlet loss

Frictional loss

Expansion loss due to the exit of the 10-inch SRV line for the 28-inch downcomers only

Exit loss

The inlet component of the loss coefficient is broken into three parts: an inlet loss due to the effects of the deflector plate, the protrusion of downcomer above the floor neglecting the vertical stiffener plates, and the effects of the stiffener plates.

Loss Due to:	24-inch	28-inch
Deflector plate (Ref. 2, diagram 3-8, P. 97)	0.28	0.14
Protrusion above floor (Ref. 2, diagram 3-1, P. 92)	0.60	0.81
Stiffener plates (Ref. 2, diagram 3-18, P. 107)	0.50	0.70
	1.38	1.65

The frictional component of the downcomer loss coefficient is given by the following equation:

$$K_f = \frac{fl}{Deq}.$$

where:

f = Darcey friction factor

L = length of pipe

Deq = equivalent pipe diameter

For the 24-inch downcomers:

f = 0.011 (Ref. 3, P. A-23)

L = 45.27 feet

Deq = inside diameter = 23.25 inches

$$K_{24f} = \frac{fl}{Deq} = \frac{(0.011)\ (45.27)}{(23.25/12)} = 0.26$$

The 28-inch downcomers must be broken down into two parts due to the ten-inch SRV line.

a. Segment containing 10-inch SRV line.

f = 0.011 (Ref. 3, P. A-23)

L = 8.17 feet

Deq = hydraulic diameter = 16.5 inches

$$K_{28fl'} = \frac{(0.011)\ (8.17)}{(16.5/12)} = 0.07 \text{ based on area} = 3.42 \text{ sq. ft.}$$

$$K_{28fl} = K_{28fl'} \quad \frac{4.05}{3.42}^2 = 0.07 \quad \frac{4.05}{3.42}^2 = 0.10 \text{ based on area} = 4.05 \text{ sq. ft.}$$

b. Segment without 10-inch SRV line.

$f = 0.011$ (Ref. 3, P. A-23)

$L = 45.27 - 8.17 = 37.1$ feet

D_{eq} = inside diameter = 27.25 inches

$$K_{28f2} = \frac{(0.011)(37.1)}{(27.25/12)} = 0.18$$

The total frictional loss coefficient for the 28-inch downcomer is the sum of the two components.

$$K_{28f} = K_{28f1} + K_{28f2} = 0.10 + 0.18 = 0.28$$

The expansion loss due to the exit of the 10-inch SRV line is computed as follows:

$A_1 = 3.42$ sq. ft $\qquad A_2 = 4.05$ sq. ft.

$$K_{exp} = \frac{(1-A_1/A_2)^2}{(A_1/A_2)^2} = 0.034 \quad \text{(Ref. 3, P. A-26)}$$

The exit loss of the downcomers is simply a sudden expansion. The maximum loss from a sudden expansion is 1.0 (Ref. 3, P. A-29) for both the 24-inch and 28-inch downcomers.

The total loss coefficients for the 24-inch and 28-inch downcomers are:

	24-inch	28-inch
K_{inlet}	1.38	1.65
K_{frict}	0.26	0.28
K_{exp}	----	0.03
K_{exit}	1.00	1.00
K_{total}	2.64	2.96

A single downcomer loss coefficient for the entire vent system was determined by taking a weighted average of the total loss coefficients for the 24-inch and 28-inch downcomers based on a single vent area. The total vent area for 99 downcomers is 309.5 square feet or 3.126 square feet/downcomer.

$$K = \frac{N_{24}}{N_t}(K_{24})\ \frac{A}{A_{24}}\ 2 + \frac{N_{28}}{N^t}(K_{28})\ \frac{A}{A_{28}}\ 2$$

K = weighted average single downcomer loss coefficient for entire vent system

$N_{24} = 83$ $K_{24} = 2.64$ $N_{28} = 16$ $K_{28} = 2.96$ $N_t = 99$

$A_{24} = 2.948$ Sq. Ft. $A_{28} = 4.05$ Sq. Ft. $A = 3.126$ Sq. Ft.

$$K = \frac{83}{99}(2.64)\ \frac{3.126}{2.948}\ 2 + \frac{16}{99}(2.96)\ \frac{3.126}{4.05}\ 2 = 2.77$$

Since the K value of 2.77 is more than thirty percent higher than the value of 1.9 presented in the analysis used in Table 6.2-1 of the FSAR, the affect on peak drywell pressure was investigated using CONTEMPT-LT (Reference 4). The input parameters for the CONTEMPT computer program were determined from the FSAR (Reference 1) with the exception of the downcomer loss coefficient of 2.77 presented above. With K = 2.77, CONTEMPT calculated a peak drywell pressure of 34.77 psig which is essentially the same as the peak drywell pressure of 34.7 psig presented in Table 6.2-1 of the FSAR.

No known studies have been performed to experimentally determine 4T test downcomer vent loss coefficients. However, in their Pool Swell Analytical Model (PSAM)/4T test data comparisons (References 5 and 6), General Electric used downcomer vent loss coefficients of 2.51 and 3.50 for the 4T test 20-inch downcomers and 24-four inch downcomers, respectively. These values were used as input to the General Electric PSAM and were calculated using information from Reference 2.

The WNP-2 downcomer friction loss coefficient (fL/D) that is used in pool swell studies is equal to 1.9 (see Table 3.8-1). Use of a value of 1.9 versus a 4T value ensures conservatism in WNP 2 pool swell studies in that lower values of fL/D maximizes pool swell velocity (see Figure 4-24 of Reference 7).

REFERENCES:

1) "WPPSS Nuclear Project No. 2, Final Safety Analysis Report", Washington Public Power Supply System, Chapter 6.2.

2) AEC-TR-6630, "Handbook of Hydraulic Resistance - Coefficients of Local Resistance And Friction", I.E. Idel'chik, 1960.

3) "Flow of Fluids Through Valves, Fittings, and Pipe", Technical Paper No. 410, Crane Company, 1980.

4) "CONTEMPT-LT--A Computer Program For Predicting Containment Pressure-Temperature Response to A Loss-of-Coolant Accident", Aerojet Nuclear Company, June 1975.

5) NEDE-21544-P, December 1976, "Mark II Pressure Suppression Containment Systems: An Analytical Model of the Pool Swell Phenomenon."

6) Response to NRC Question 020.071, transmitted via letter MFN-275-78 to Mr. J.F. Stolz, Chief, Light Water Reactor Branch No. 1, USNRC, from Mr. L.J. Sobon, Manager BWR Containment Licensing, General Electric Company on "Responses to NRC Request for Additional Information (Round 3 Questions)", dated June 30, 1978.

7) NEDE-21061 Revision 3, Mark II Containment Dynamic Forcing Functions Information Report, June 1978.

Q 22.032

Provide the following information regarding the vacuum breaker systems between the wetwell and the drywell and between the reactor building and the wetwell:

a. Describe the preoperational and inservice tests that will be performed to verify that the set-points of the vacuum breakers are at the appropriate pressure levels and meet the required opening times.

b. Provide the sensitivity limits and hysteresis characteristics of the electrical switches. Provide the results of your analyses of the maximum opening between the valve disc and the seat when the position indicator system indicates that the wetwell vacuum breaker valve is closed.

c. Provide a schematic of the vacuum breaker assembly. Provide your analysis of the minimum flow area and the total loss coefficient for one vacuum breaker assembly.

Response:

a. Preoperational and inservice testing of the vacuum breaker system is performed to verify that the valves open at the appropriate pressure levels. For the single and double disk check valves, testing will be accomplished using a torque wrench applied to the disk pivot shaft and determining the opening pressure by correlation with the measured torque required to open the valve. Correlation curves are provided by the valve manufacturer. The response time for opening of the single and double disk check valves is not measured during pre-operational and inservice testing. Two of the double disk valves were tested by the manufacturer for compliance with the specification requirement that the valves be fully opened within 1 second at 0.5 psi differential pressure.

b. The switches used for position indication of the wetwell-drywell and reactor building wetwell vacuum breakers are a contact-probe type. These contact probes are very sensitive and have zero hysteresis. Accuracy within 0.010" is possible.

Valve closed indication is taken directly from the valve face. Four probes are located 90° apart, straddling the valve vertical centerline. Due to the accuracy of the switches (0.010"), the location of the four probes, and based on the geometry of the vacuum breaker, the maximum opening between the valve disc and the seal when the position indicator system indicates a closed valve is 0.012".

c. The attached Figures (2-1 and 2-2) from the manufacturer's instruction manual illustrate the configuration of the double disk vacuum breaker assembly, and the seal detail. FSAR Reference 3.8-8, which was provided to the NRC, also provides illustrations and details of the vacuum breaker system and the Anderson, Greenwood & Co. vacuum breaker valves.

The minimum flow area of the vacuum breaker valves is 295.6 square inches based on the 19.4 inch diameter inlet orifice. Capacity certification tests were performed on 3 wetwell-drywell and 3 reactor building-wetwell vacuum breaker valves. Using the flow data from these tests, resistance coefficients (K) were determined as follows:

Wetwell-Drywell valves - 4.73 at 0.360 psid

Reactor Building-Wetwell valves - 1.65 at 0.289 psid

These values of K include valve entrance effects and are based on the connecting piping internal area of 424.6 square inches.

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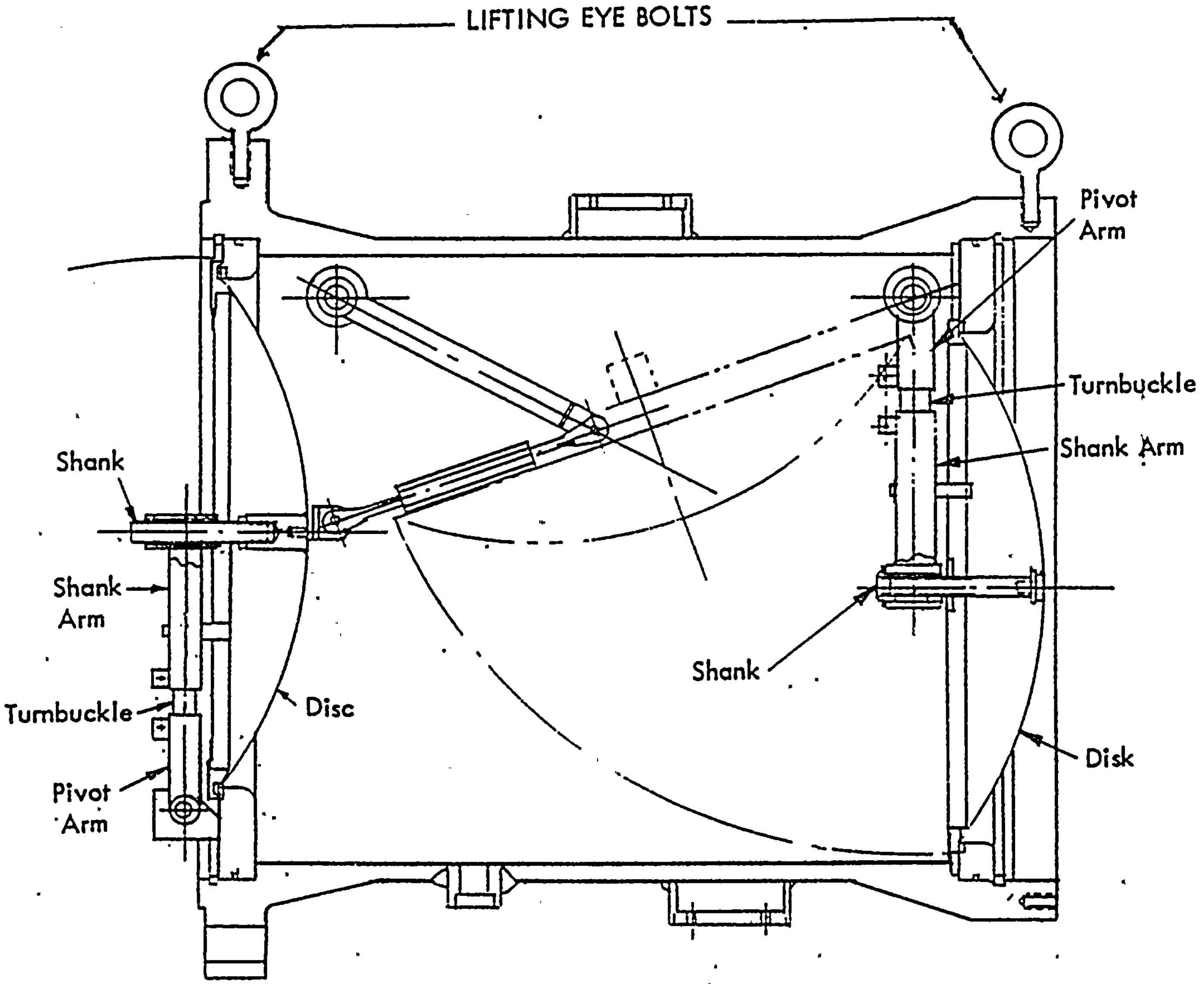
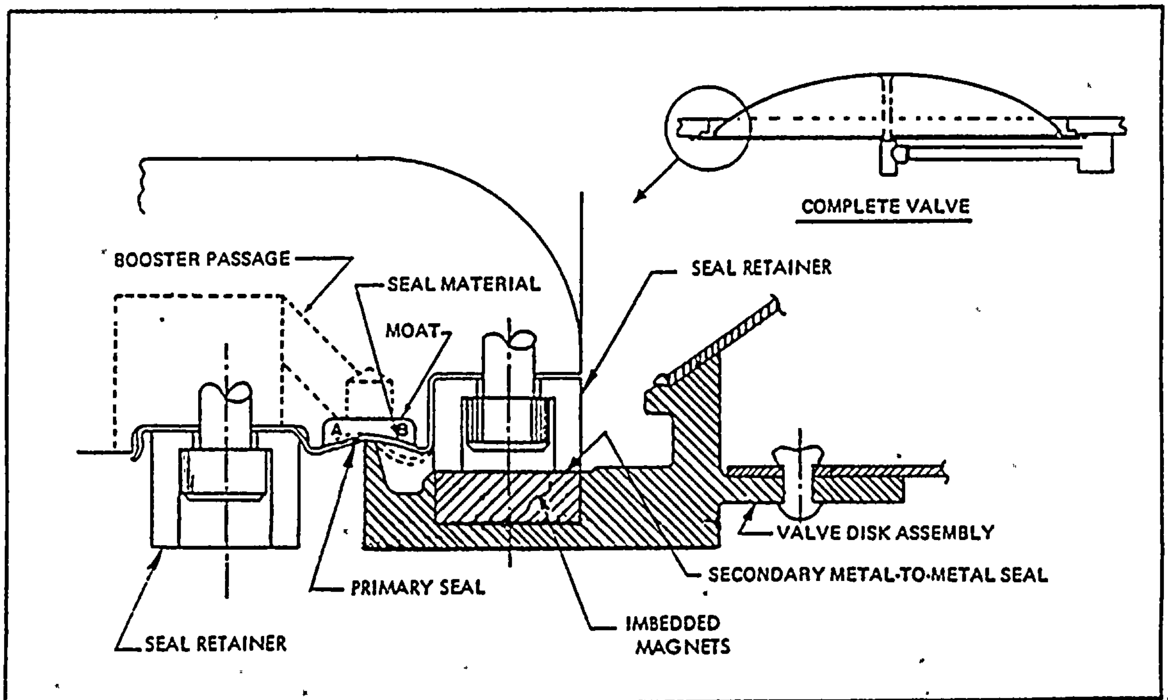


WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	VALVE CROSS SECTION (DUAL DISC)	FIGURE 2-1.

2-4. VALVE SEAL. A circular seal around the perimeter of the valve orifice provides a superior seal when the valve is in the closed position. The details of this seal are shown in Figure 2-2. Use the small sketch labeled COMPLETE VALVE for orientation, then refer to the CV1-L seal detail. The valve disk assembly (cross hatched) is the movable part. All other parts are fixed. A circular moat encircles the valve orifice. Over this moat the valve seal diaphragm material is stretched and held in place by two concentric retainer rings. The valve seals when the seal lip on the valve disk assembly touches this diaphragm. The inner of the two seal retainer rings serves to limit the penetration of the lip into the seal from 0.010 to about 0.020 inches and also serves as a secondary metal-to-metal seal. The primary seal between the lip and diaphragm is pressure boosted. Pressure from the high pressure, or downstream side is fed into the moat under the seal diaphragm. This causes a zero pressure difference across the seal at point A and a full pressure difference across the seal diaphragm at point B.



Q. 022.033

You state in 6.2.1.1.8.2 of the FSAR that operation of the containment purge system will be limited to one percent of the reactor operating time. We find this approach acceptable provided you:

a. Expand your definition of reactor operating time to include the three operational modes of start-up, hot standby, and hot shutdown.

b. Demonstrate that the purge system isolation valves can be closed when subjected to the environmental conditions, including pressure, that occur following a postulated loss-of-coolant accident.

c. Combine the time to purge the suppression chamber with the time to purge the drywell in the proposed one percent restriction on the operating time of the containment purge system.

Response:

a. Refer to response to question 022.019.

b. Refer to response to question 022.019.

c. Purge system operation, addressed in paragraph 6.2.1.1.8.2, includes purging the drywell and suppression chamber.

Q. 22.034

Identify all access openings to the secondary containment and discuss the administrative controls that will be exercised over them. Discuss the instrumentation to be provided to monitor the status of the openings. Indicate whether position indicators will provide readout information to the plant operator and whether alarms will be annunciated in the main control room.

Response:

All access openings to the secondary containment are administratively controlled. Details related to this administration and the indications provided are presented in the WNP-2 Security Plan, transmitted under separate cover to the NRC.

Q. 022.035

Provide the following information related to potential bypass leakage paths:

a. For each air or water seal, perform an analysis of the fluid inventory which will be available to maintain the seal for 30 days following a postulated loss-of-coolant accident and demonstrate that this fluid inventory will be sufficient. Describe the testing program and the specific details of your proposed technical specifications which will verify the assumptions used in the analysis. Provide the basis for the valve fluid leakage used in your analysis.

b. For each of these paths where water seals eliminate the potential for bypass leakage, provide a sketch showing the location of the water seals relative to the system isolation valves

c. Explain why the combustible gas control system is omitted from Table 6.2-13 of the FSAR as a potential leakage path. Demonstrate that this system meets each of the provisions of Branch Technical Position CSB 6-3, Section B-9, for a closed system.

Response:

a. Potential bypass leakage paths around the secondary containment are discussed in 6.2.3, Secondary Containment Functional Design. As discussed in 6.2.3.2, the two 24-inch reactor feedwater (RFW) lines are the only lines for which a water or air seal is assumed which prevents secondary containment bypass leakage. A vertical water seal of about 50 feet normally exists between the inboard isolation check valve and the RPV nozzle for each feedwater line (see Figure 6.2-25). During a LOCA, flashing due to depressurization will blow out most of the water seal in the line. However, several feet of water will remain to provide a temporary barrier against back leakage through the RFW line.

Long-term (greater than 30 days) protection against back leakage will be accomplished by closing the motor-operated gate valve in each RFW line. This remote-manual gate valve is located just outboard of the two isolation check valves (one inboard and one outboard of containment) in each RFW line. The gate valve may be shut subsequent to a LOCA any time the operator determines that RFW flow is unnecessary or unavailable. The gate valve provides a third barrier against back leakage and precludes any credible bypass leakage through the RFW line. For some accidents, feedwater may be available using the condensate and condensate booster pumps.

As discussed in 6.2.3.3 and 6.2.6.3, the isolation valves on the lines which were identified as potential bypass leakage paths around the secondary containment, will be tested to ensure that the individual leakage rates are below the limits allowed by ASME Code Section XI, IWV-3426. The limit allowed by IWV-3426 was the value assumed in calculating the water lost from the water seal through the RFW isolation valves.

b. Figure 6.2-25 shows the RFW line routing.

c. The containment atmosphere control (CAC) system is a closed system outside the primary containment. Suction and discharge are to the primary containment. All piping remains within the secondary containment. Any leakage from the CAC system will be processed by the standby gas teatment system prior to release to the environment. The CAC system is described in detail in 6.2.5 and shown in Figure 3.2-17.

The CAC system meets all the criteria stipulated in BTP CSB 6-3 paragraph B.9. The CAC system does not directly communicate with the environment, is designed to Code Group B standards, meet Seismic I design requirements, is designed to the primary containment pressure and temperature design conditions, is designed against the consequences of any breach in the reactor coolant pressure boundary (pipe whip, etc.), and will be open to the primary containment atmosphere during the integrated leak rate test. In addition, the CAC system can be isolated from the primary containment by two, redundant isolation valves. There is no reason to consider the CAC system as a secondary containment bypass leakage path.

Q. 022.036

It is our position that safety-related equipment located in the drywell should be exposed for 6 hours to a saturated steam environment at 340°F and a pressure equal to the drywell design pressure during the qualification testing. (Refer to Item 022.021 transmitted on September 18, 1978).

Response:

Safety related equipment in the drywell for WNP-2 was procured before 1974 and was specified and qualified to a saturated steam environment for 3 hours at 340°F and 6 additional hours at 45 psig (drywell design pressure). These environmental conditions formed the design basis for WNP-2 in the drywell and were consistent with IEEE standard 323-1974 (Table A2), "Test Conditions for Boiling Water Reactors" when it was published.

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Q. 022.036

It is our position that safety-related equipment located in the drywell should be exposed for 6 hours to a saturated steam environment at 340°F and a pressure equal to the drywell design pressure during the qualification testing. (Refer to Item 022.021 transmitted on September 18, 1978).

Response:

Safety related equipment in the drywell for WNP-2 was procured before 1974 and was specified and qualified to a saturated steam environment for 3 hours at 340°F and 6 additional hours at 45 psig (drywell design pressure). These environmental conditions formed the design basis for WNP-2 in the drywell and were consistent with IEEE standard 323-1974 (Table A2), "Test Conditions for Boiling Water Reactors" when it was published.

Q 22.037

Discuss in detail the design provisions incorporated for periodic inspection and operability testing of the individual components of the containment heat removal systems, including the pumps, valves, duct pressure-relieving devices and spray nozzles.

Response

The containment heat removal system is an operating mode of the residual heat removal (RHR) system shown in Figure 3.2-6. Operation of the RHR system for containment heat removal is discussed in 6.2.2.

All power operated valves can be exercised without affecting reactor operation. All RHR check valves in the primary containment, RHR-V-41A, B and C and RHR-V-50A and B, can be remotely exercised by means of a pneumatic actuator (see Note 3 of revised Table 6.2-16). The check valve on the discharge of each RHR pump will be exercised when the RHR pump is operated. All relief valves can be removed and bench tested when the RHR system is not required.

The RHR pumps have a full flow test line back to the suppression pool which allows testing of the pumps without affecting reactor operation. Sufficient instrumentation is available to verify proper flow, NPSH, and discharge pressure. The pumps are in an accessible area outside the primary containment where, if necessary, they can be locally monitored.

The spray nozzles on the containment spray headers are passive components. Each drywell spray header contains 150 spray nozzles. Each spray nozzle consists of a nozzle body and seven removable spray caps. Each spray cap has an internal vane. Since the spray nozzles are passive components with no moving parts, testing is limited to a qualitive air flow test during preoperational testing.

The RHR heat exchangers are periodically used for cooling down the reactor pressure vessel after shutdown in preparation for maintenance and refueling. This verifies operability of the RHR heat exchangers.

There are no duct pressure relieving devices, or anything similar, on the RHR system.

In addition, the RHR system is built to Code Group B (ASME III-2) standards and is subject to the applicable inservice inspections discussed in 6.6.

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Q 22.038

Provide a detailed analysis of the available net positive suction head for the pumps in the reactor heat removal systems that are used as part of the containment heat removal system. This analysis should demonstrate compliance with the guidance contained in Regulatory Guide 1.1, "NPSH for Emergency Core Cooling and Containment Heat Removal System Pumps." Indicate the required net positive suction head.

Response

The net positive suction head (NPSH) for all emergency core cooling system (ECCS) pumps was calculated in accordance with Regulatory Guide 1.1.

NPSH = Wetwell air space pressure + static pressure - friction losses - vapor pressure

Static head equals minimum suppression pool water level, 466 feet, minus centerline of RHR pump suction nozzles, 421 feet. Static head equals 45 feet.

Friction losses for suction piping are less than 3 feet for all RHR pumps. The suction is assumed to be 50% plugged.

Vapor pressure at the peak suppression pool temperature of 220°F is 2.5 psig (6 feet).

In accordance with Regulatory Guide 1.1, "no increase in containment pressure from that present prior to postulated loss-of-coolant accidents" is assumed. Therefore, the wetwell air space pressure is assumed to be 0 psig, even though maximum suppression pool temperature is 220°F. This is conservative but not realistic since the suppression pool will be at saturation pressure any time the suppression pool water exceeds 212°F.

Based on the above, the NPSH available is 36 feet. The NPSH required by the pump manufacturers as documented by pump performance curves is 11 feet at 7450 gpm rated flow. See figure 6.3-10a, b, c for pump performance curves.

Q. 022.039

Describe the analysis performed to establish the size of the suction screens in the reactor heat removal system. Provide a drawing showing the suction screen assembly.

Response:

The screen size for the suction strainers on the residual heat removal system is based on the more restrictive criteria set by the pump manufacturer or the spray nozzle orifice opening.

The pump manufacturer imposed a maximum particle size of 0.09375 inch, based on the size of the smallest orifice/flow path in the pump mechanical seal. This is significantly more restrictive than the requirement imposed by the spray nozzles which have an orifice opening of 0.26563 inch. Accordingly, the strainers have been specified to prevent the passage of particles 0.09375 inch or greater.

The suction strainers (2 per pump) have been procured to the following specifications:

Primary Service Rating: ANSI 1501-1

Quality Class I

Seismic Category I

Cleanliness Class B

Applicable Code: Strainer materials and fabrication shall meet ASME Section III, Class 2 requirements. The "N" stamp shall not be applied since the strainers cannot be hydrostatically tested.

Materials: Strainer body shall be stainless steel 304 or 316, or engineer approved equal, suitable for submergence in high quality water during a 40-year lifetime.

The strainers are cylindrical, as shown on Figure 022.039-1. Strainer hole diameter is 0.09375 inch. Strainers are attached to ANSI 150# RF Flanges.

Quantity	Nominal Diameter (inches)	Maximum Length (inches)	Rated Flow (per strainer) gpm
10	24	22	4200
2	8	14	300
2	6	12	288

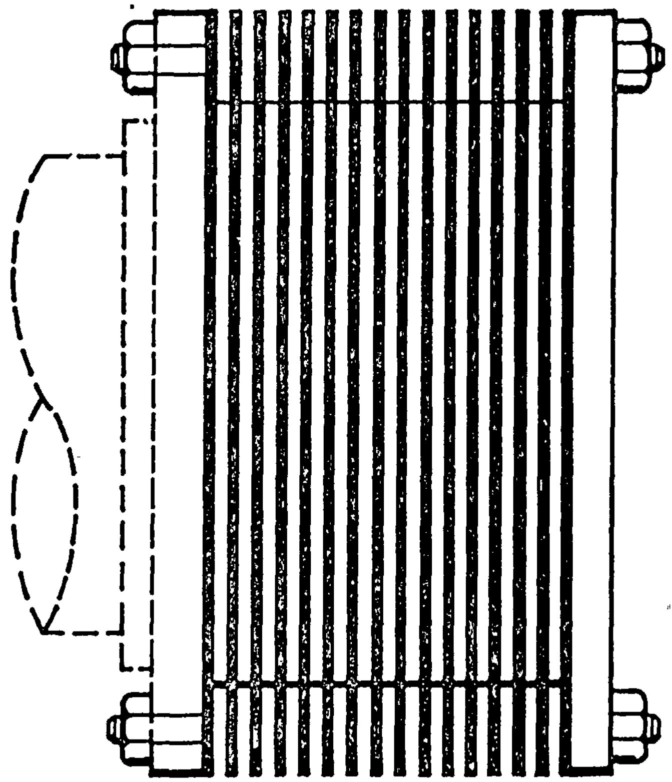
Head loss shall be limited to four (4) feet of water assuming the strainer 50% clogged and a water temperature of 220°F.

Acceleration loads have been developed in concert with the Mark II hydrodynamic load program. They have been applied concurrently with the load due to process flow through the 50% clogged strainer and a 25 psid (equivalent static) load applied across the projected area of the strainer which results in the most limiting total loading.

The drawing of a typical strainer is provided in Figure 022.039-1.

Amendment No. 30
June 1983

MEASUREMENTS FOR STRAINERS AT
PENETRATIONS X-32, X-35, X-36
RATED FLOW: 8,400 gal/min



NOTES:

1. *FLOW STATED ABOVE IS PER PENETRATION WITH TWO (2) UNITS DESCRIBED ABOVE REQUIRED PER PENETRATION.*
2. *UNITS ARE DESIGNED, MANUFACTURED AND INSPECTED IN ACCORDANCE WITH ASME SECTION III, CLASS 2 (NOT STAMPED) 1974 ED. WITH ADDENDA THRU WINTER 1976*
3. *DESIGN TEMP: 220°F*

WASHINGTON PUBLIC POWER **SUPPLY SYSTEM** NUCLEAR PROJECT NO. 2	TYPICAL STRAINER	FIGURE 022.039-1

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Q 22.040

Provide a full scale drawing for Figures 3.2-2, 3.2-3, and 3.2-6 of the FSAR.

Response

Enclosed are seven copies each of the requested full scale drawings.

Q. 022.041

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Q. 022.042

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Q. 022.043

DELETED

Q. 022.043

DELETED

Q. 022.47

You state in 6.2.5.1 of the FSAR that purging hydrogen from the containment is not required as a backup system to the hydrogen recombiners since all components of the recombiners are redundant. We have provided guidance in Section C.4 of Regulatory Guide 1.7 which states that the capability of a controlled purge of the containment atmosphere should be provided to aid in cleanup. Discuss your plans to comply with the guidance contained in Regulatory Guide 1.7.

Response:

Refer to 6.2.1.1.8.3, 6.2.5.1.m, and 6.2.5.2.4.

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Q 22.048

You state in 6.2.5.3.1.3 of the FSAR that the corrosion of aluminum, zinc, and zinc base paints located either in the drywell or in the suppression chamber were determined to be insignificant. However, we have determined that a potential hydrogen release from the corrosion of zinc following a postulated loss-of-coolant accident should be considered in the analysis of the total hydrogen production and accumulation within the containment. Accordingly, provide the following information:

a. Provide the corrosion rate as a function of temperature for all materials in the containment that could become a source of hydrogen due to corrosion.

b. Describe how the corrosion rates assumed for the materials identified in Item (a) were established: Identify the experimental data base, including the appropriate references, and discuss the conservatism in the applicability of the data in view of the calculated environmental conditions following a postulated loss-of-coolant accident.

c. Provide the mass and surface area of zinc paint and galvanized steel and other corrodible materials in both the drywell and the wetwell.

d. Provide a graphic representation of the total hydrogen concentration inside the containment as a function of time with (1) no hydrogen recombiners operating; (2) one recombiner operating; and (3) both recombiners operating.

e. Provide a graphic representation of the contribution of each source of hydrogen as a function of time.

f. Describe the periodic surveillance that will be done to demonstrate the operability of the hydrogen recombiners and the backup purge system.

g. Identify the location of (1) the hydrogen sample points in the drywell and the suppression chamber, and (2) the suction and discharge points of the combustible gas control system with respect to nearby structures and equipment.

Response:

WNP-2 is an oxygen control plant with catalytic hydrogen/oxygen recombiners. Postulating low rates of hydrogen generation is conservative for determining maximum containment oxygen concentration post LOCA since the availability of additional hydrogen would increase the removal rate of the oxygen. For this reason, the only sources of hydrogen used to determine containment oxygen levels are from the fuel clad metal-water reaction and from radiolysis. Additional sources such as the zinc rich paint-water reaction are only assumed to determine the maximum catalyst temperature and the hydrogen level is assumed to be at least twice the oxygen level for this analysis. Since the maximum hydrogen concentration achievable in containment following a LOCA is not a concern as long as oxygen levels are maintained below flammability limits, detailed analysis with conservative assumptions on the hydrogen generation rates and methods of production are not required. Section 6.2.5 of the FSAR discusses the hydrogen generation rates for the metal-water reaction and radiolysis and refers to figures that show containment concentrations vs. time for various system configurations.

The location of the hydrogen and oxygen sample points in the drywell and suppression chamber with respect to nearby structures and equipment has been answered in response to Question 022.025.

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WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PLANT 2	INTEGRATED HYDROGEN PRODUCTION AS A FUNCTION OF TIME (WITHOUT HYDROGEN RECOMBINING)	FIGURE 022 .048-2
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Q 22.050

Those closed systems outside containment which must function following an accident become extensions of the containment boundary after a postulated loss-of-coolant accident. Certain of these systems may also be identified as one of the redundant containment isolation barriers. Since these systems may circulate water or the containment atmosphere which may be contaminated, components of these particular systems which may leak are relied on to provide containment integrity. Accordingly, provide the proposed leakage limit for each system that becomes an extension of the containment boundary following a postulated loss-of-coolant accident. Discuss your plans for leak testing the systems either hydrostatically or pneumatically. Discuss how the process leakage limits will be included in the radiological assessment of the site.

Response:

Closed systems which become an extension of the primary containment during the post-LOCA period are the:

High Pressure Core Spray (HPCS)
Low Pressure Core Spray (LPCS)
Residual Heat Removal (RHR)
Containment Atmosphere Control (CAC)
Reactor Core Isolation Coolant (RCIC)*

The above systems will be given a hydrostatic or pneumatic test following system completion per Section III of the ASME Code. This test will be repeated once every ten years per the requirements of Section XI of the ASME Code. In addition to these tests, the HPCS, LPCS, RHR and RCIC (water side) systems are maintained under constant pressure. These systems are kept full of water and pressurized by either water leg pumps or by the suppression pool static head. During normal operation any significant leakage from valve packing or pump seals in these systems would be detectable by visual observation or by sump level alarms. Also the pumps will be flow tested quarterly and the valves will be exercised per the ASME, Section XI inspection program. During the tests a visual inspection for water leakage from the component will be made. In addition, all containment isolation valves in the above systems subject to reactor operating pressure are monitored during normal operation for valve stem leakage by the valve packing detection system (see Figure 3.2-6 (Note 5), 7, and 8 (Note 4)).

* The RCIC system is included here as it is a closed system outside containment and may briefly operate following a LOCA. It is not required to operate post-LOCA, however, and will be isolated once reactor pressure is reduced.

Defining the threshold value at which leakage from a component would become visible is difficult, however the value can realistically be assumed to be below .1 gpm. Based on this and the fact that excess leakage paths would be repaired, any water leakage during the post-LOCA period is expected to be negligible.

The RCIC steam lines from the reactor to the inside containment isolation valve and the CAC system will be open to the primary containment atmosphere during the Type A test. Leakage from these lines will be part of the measured allowable leakage for the Type A test.

All these systems are entirely within the secondary containment and consequently any leakage during the post-LOCA period will be processed by the standby gas treatment system (SGTS). Appendix B to the Standard Review Plan 15-6.5 deals with leakage from ESF systems during a LOCA and states "when ESF - grade filters are supplied, no doses resulting from passive failures need be considered." Since the SGTS includes ESF - grade filters these doses have not been included in the site radiological assessment total LOCA dose figure. However, an analysis has been done to confirm the fact that the doses resulting from passive failures such as minor leaks in ESF systems is negligible and is outlined in the response to question 312.12.

Q. 022.053

The Caorso test results discussed in your report on the safety/relief valve (SRV) loads, "SRV Loads-Improved Definition and Application Methodology for Mark II Containments," exhibit some high frequency pressure spikes in the boundary pressure measurements during the initial phase of the air clearing transient. Since these spikes were not observed in previous quencher test data (i.e., the German quencher tests), this phenomenon suggests that the specific type of quencher design may be important in determining the characteristics of air clearing loads. Accordingly, provide a detailed description of the WNP-2 quencher geometry, including a description of the hub design. Compare the geometry of the WNP-2 quencher design with the device tested in Caorso. Discuss any differences that may exist between them. Indicate how these differences might influence air clearing loads.

Response:

As can be seen in Table 022.053-1, the geometric properties of the quencher devices used in WNP-2 and Caorso are essentially identical. The arms are of the same diameter and thickness and differ in length by only 3/4". The holes in the quencher arms are identical in number, size, spacing, orientation, and location with respect to the center line of the quencher. The hubs are of the same size for both quenchers. WNP-2 has a slightly thicker quencher hub than Caorso. The conical transition piece from SRVDL to quencher hub has a slightly more gradual taper for WNP-2 as compared to Caorso (10° vs. 13.5°). Since the quencher configurations are nearly identical as outlined above, there is no reason to expect a difference in the characteristics of air clearing loads.

The only geometric difference in the plan orientation of the quencher arms is illustrated in Figure 022.053-1. WNP-2 quencher arms form two central angles of 80° and two central angles of 100°. At Caorso the quencher arms form three central angles of 80° and one central angle of 120°. This results in two of four arms being shifted 20° with respect to Caorso quencher arms. This minor difference in arm orientation should have no effect on air clearing loads.

The SRV Report defines two types of time histories for the forcing function: one which exhibits some initial high frequency pressure spikes and another which looks more like a typical single frequency wave form (claimed to be observed during German quencher tests). Both types were observed during Caorso tests. Consequently, since the WNP-2 design basis time history specifies use of both, this should take care of the specified concern.

TABLE 022.053-1

COMPARISON OF QUENCHER GEOMETRY

	WNP-2	Caorso
Number of Arms	4	4
Length of Arms (from Hub)	4'-11-1/4"	4'-10-1/2"
Diameter & Thickness of Arms	12" sch. 80	12" sch. 80
Number of Holes per Arm	1496	1496
Size of Holes ( )	0.39"	0.39"
Spacing of Holes	1.96"	1.96"
Quencher to 1st Row of Holes	1"-10-3/4"	1'-10-3/4"
Hub Size ( )	24"	24"
Hub Thickness	2.3"	2.0"

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WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	QUENCHER ARM ORIENTATION	FIGURE 022. 053-1

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Q. 022.054

The Caorso test results discussed in the SRV report cited above, indicate that the size of the vacuum breaker on the SRV line is important in determining the reflood transient after valve closure and, consequently, the subsequent valve actuation loads. Indicate whether the size, number and characteristics of the vacuum breakers installed on the SRV lines of the WNP-2 facility are similar to those of the Caorso plant. If there are differences, discuss what effects you expect these differences may have on your facility. State how these differences and their effects will be incorporated in your load definition for the WNP-2 facility.

Response:

Two vacuum breakers in parallel are installed on each of the 18 main steam relief valve discharge lines in WNP-2. In designing the vacuum breakers, the effects of the sizing, opening time, pressure losses, and delta P were considered for each discharge line. From these considerations, a set of conservative characteristics necessary to provide protection for the discharge lines for subsequent actuation was determined. A comparison of the characteristics of the WNP-2 vacuum breakers with those in the Caorso plant is as follows:

	WNP-2	Caorso
Manufacturer	GPE Controls	Atwood & Morrill Co.
Size (in.)	10	10
Type	Single Wafer type with swinging disk	Single straight through with swinging disk
Number	18	16
Flow area (in^2)	38.48	78.54
Design Conditions:		
Flow	1966 SCFM*	14000 CFM
Pressure (psig)	-0.5 to 2	0.0
Delta P (psid)	0.115	7
Set point (psid)	0.1	Not available
Opening time (sec. at psid)	0.21 @ 0.4	Not available
$A/\sqrt{K}$ (ft^2)	0.278(40 in^2)	0.72(104 in^2)

* Acceptance point at 0.115 psid.

Differences between the values listed are apparent; the flow area, design flow, delta P, and $A/\sqrt{K}$ of the Caorso vacuum breakers are larger. This arises because they are expressed in terms of quantities occurring at different operating points of their respective operating characteristic curves. The WNP-2 vacuum breakers, for example, are expressed in terms of values occurring at delta P = 0.115 psid, whereas those in Caorso are expressed in terms of values occurring at delta P = 7 psid. At these unequal differential pressures, the flows, pressure losses ($\sqrt{K}$), and area-resistance coefficients ($A/\sqrt{K}$) are expected to be different. If the 14,000 cfm flow through the Caorso vacuum breakers is scaled down by the square root of the ratio 0.115 psid/7 psid (since the flow through the vacuum breakers varies approximately as the square root of delta P), the Caorso vacuum breakers are calculated to pass approximately the same equivalent flow as the WNP-2 vacuum breakers. As a result, the two vacuum breakers should behave in essentially the same manner.

As has been indicated, vacuum breakers determine the reflood transient within the SRV line. The reflood transient establishes the initial water level condition within the SRV line prior to an SRV discharge event, i.e., an initial condition for the event. Test data for a diversity of initial SRV line water level conditions (low, normal, and high water level),* were gathered during the Caorso tests. The SRV load definition for WNP-2 envelopes the loads observed at Caorso. As such, differences which might exist in vacuum breakers are accounted for by this bounding approach to load definition which envelopes test data obtained for low, normal, and high water level initial conditions.

*See Reports: NEDE-25100-P and NEDE-24757-P, Caorso Phase I and II test reports.

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Q. 022.055

The design values for the transient SRV loads in the WNP-2 facility are based on single valve, subsequent actuation data from Caorso in-plant tests. These design values are then used in load cases involving multiple valve actuations based on the assumption that actuation of multiple valves occurs only for the first actuation of the SRVs. State whether this assumption can be supported by a transient analysis of the worst transient event expected in the WNP-2 facility. If this is not the case, revise your load definition to consider the multiple valve effect on the design basis pool boundary loads. Our concern is that the Caorso test results indicate that pressure loads from multiple-valve actuations are greater than those from single-valve actuations under similar first actuation conditions.

Response:

The SRV report states that a multiple valve actuation case is more likely to be an initial actuation rather than a subsequent actuation. This statement is only made to show added conservatism in the design value since initial actuations are expected to be lower in peak pressure amplitude than subsequent actuations.

There is insufficient data from the Caorso tests to provide a statistical base for multiple valve actuations. For comparison purposes though the maximum measured pressures at P19 may be examined:

P_{max} for single valve (valve A), initial actuations = 5.96 psi (See Table 4.1 of SRV report)

P_{max} for multiple valve, initial actuations = 5.87 psi (See Table 4.5 of SRV report)

If the initial actuation tests measured at Caorso are used as the data base for comparison purposes, it is found that a single valve test has recorded the highest pressure amplitude.

If all of the Caorso tests are used as the data base, it is again found that a single valve test has recorded the highest pressure amplitude.

Furthermore, in comparing design envelope frequency spectra with multiple valve actuation data from the Caorso tests (see Question 022.056, Figure 022.056-2), conservative results are observed.

It was concluded that since single valve actuations of Caorso tests yields higher maximum pressure amplitudes and since the frequency spectra envelope of all multiple valve tests performed at Caorso is enveloped by the design envelope frequency spectra that the load definition is indeed adequate.

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Q. 022.056

In Figure 6.8 of the SRV report cited above, you indicate that the frequency spectrum of the design pressure-time histories can bound the experimental pressure-time traces at a statistical confidence level of 90 percent/90 percent and also bounds the envelope of the single valve subsequent actuation pressure traces from the Caorso tests. Provide similar comparisons for leaky valve (LV) first actuation data and for multiple valve actuation (MVA) data. Our concern is that the distinct differences in the characteristic in LV data (e.g., the frequency and amplitude) and the greater number of initial pressure spikes in MVA data.

Response:

a. Comparison of Design Envelope Versus Leaky Valve Actuation Data

A comparison between the design envelope response spectra and the leaky valve first actuation response spectra (measured at P19) is shown in Figure 022.056-1. The design curve completely envelopes the leaky valve first actuation envelope recorded from Caorso data thereby justifying the design curve as adequate. The leaky valve first actuation envelope is developed from tests, 41, 42, 43, and 44 of Caorso Phase II tests.

b. Comparison of Design Envelope Versus Multiple Valve Actuation Data

In the case of single valve actuations, valve "A" was considered and, as shown in the SRV reports, data recorded by sensor P19 was considered adequate to represent the local as well as the global boundary pressure frequency content. In the case of multiple valve actuation, in order to obtain a valid comparison with the design (global) pressure, one must determine a global (averaged) frequency content about the quenchers which are actuated. Pressure sensors corresponding approximately to the position of P19 in reference to valve "A" are selected for each of the actuated quenchers. Their readings could be used as a measure of local boundary pressure frequency content. However, their average is used for comparison with the design (global) pressure. Caorso tests were instrumented to record local data for quenchers "A"

(sensor P19) "E" (sensor P50) and "U" (sensor P51). Therefore, for each test in which quenchers A, E, and/or U are actuated, the locally recorded frequency spectra are averaged to give a measure of global response for that test. An averaged frequency spectrum envelope is then generated for the multiple valve actuation tests considered, and is compared to the design envelope as shown in Figure 022.056-2 with conservative results. The multiple valve actuations frequency spectrum envelope is developed from tests 24-27, 29, 30, 32, 45-1, and 45-2.

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WASHINGTON PUBLIC POWER SUPPLY SYSTEM **NUCLEAR PROJECT NO. 2**	DESIGN ENVELOPE RESPONSE SPECTRUM VS. LEAKY VALVE INITIAL ACTUATION RESPONSE SPECTRUM ENVELOPE	FIGURE 022. 056-1

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WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	DESIGN ENVELOPE FREQUENCY SPECTRUM VS. MULTIPLE VALVE ACTUATION FREQUENCY SPECTRUM ENVELOPE (AVG.)	FIGURE 022. 056-2

Q. 022.057

Many of the Caorso subsequent actuation tests were conducted with one of the two vacuum breakers blocked. You used the results from these particular tests to derive the design values of SRV pressure transients for the WNP-2 facility. However, the maximum pool boundary pressure measured in the Caorso tests is from a subsequent actuation test with both vacuum breakers operating (Test 22A02), which we believe to be prototypical for Mark II plants. The maximum measured value of the peak positive pressure is 8.7 psi and the mean value is 5.9 psi for subsequent actuation tests with only one vacuum breaker functioning. They are 9.4 psi and 7.3 psi, respectively, when two vacuum breakers are functioning. This represents a potential nonconservatism in the data base used in the derivation of design values of SRV pressure transients for the WNP-2 facility. Accordingly, discuss this phenomenon and its effect on the data evaluation, including your derivation of the design basis SRV loads.

Response:

As indicated in the response to Question 022.054, vacuum breakers affect SRV loads by influencing the reflood transient which in turn establishes initial SRV line water level, i.e., the vacuum breakers influence one of the initial conditions for SRV discharge. The Caorso test matrix includes few subsequent actuations with both vacuum breakers operating; too few actuations to make any statistical conclusions. However, the maximum pressure amplitude is of interest. The P 90/90 value for Caorso data equals 9.37 psi.

The maximum pressure amplitude recorded at Caorso for subsequent actuations for cases involving the operation of either one or two vacuum breakers is 9.4 psi which is equivalent to the statistically derived P 90/90 value.

It can, therefore, be concluded that the approach used in the development of a design pressure amplitude for WNP-2 is indeed a justified and conservative approach since the P 90/90 pressure amplitude used in the determination of a WNP-2 design pressure amplitude is not exceeded by any pressure amplitude measured during the Caorso tests. For further discussion, see the response to Question 022.054.

Q. 022.058

In order to account for the differences between the WNP-2 design conditions and the Caorso test conditions (e.g., the pool geometry, the number of SRVs and the initial pool temperature), you used a pressure amplitude multiplier based on a correlation in the Design Forcing Function Report (DFFR) to obtain the WNP-2 design values for SRV loads. This procedure involves the extrapolation of pressure amplitudes measured in the Caorso tests with respect to some parameter values (e.g., the SRV steam flow rate) to WNP-2 design conditions using the trends established in the DFFR. Accordingly, provide justification for your position that the trends used in this extrapolation can be supported by available Caorso data.

Response:

The DFFR established trends for pressure amplitude variation with different parameters such as steam flow rate, initial pool temperature, etc., are supported by available Caorso data.

As can be seen from Figure 022.058-1 the DFFR predicted pressure amplitudes increase with steam flow rate. Caorso Phase II tests #35 and #38 were selected since all test conditions, other than steam flow rate, were maintained approximately equal during these tests:

$$\left[\Delta P_{\text{predicted (DFFR)}} = 1.45 \text{ psi}\right] \approx \left[\Delta P_{\text{measured}} = 1.5 \text{ psi}\right]$$

Similarly, Figure 022.058-2 illustrates that the DFFR predicted pressure amplitude increases with initial pool temperature. This is also verified by available Caorso data. Caorso Phase I tests #7 and #1301 were selected in this case since all conditions, other than the initial pool temperature, were maintained approximately equal during these tests:

$$\left[\Delta P_{\text{predicted (DFFR)}} = 0.4 \text{ psi}\right] \approx \left[\Delta P_{\text{predicted (Caorso Phase I SVA)}} = 0.5 \text{ psi}\right].$$

It is concluded from the above examples that DFFR established trends are supported by available Caorso data and could be used to account for the differences between the WNP-2 design conditions and the Caorso test conditions.

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WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	PRESSURE - STEAM FLOW RATE TREND COMPARISON DFFR VS. CAORSO	FIGURE 022. 058-1

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WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	PRESSURE - INITIAL POOL TEMPERATURE TREND COMPARISON DFFR VS. CAORSO	FIGURE 022. 058-2

Q. 022.059

The proposed vertical pressure distribution in the SRV report cited above, is constant between the bottom of the suppression pool and the quencher elevation and then decreases linearly to zero at the pool surface. You stated that you derived this particular spatial variation by reviewing the maximum pressures measured at various elevations in the Caorso tests. However, as shown in Figure 3.8a of the cited report, this proposed pressure distribution cannot bound the maximum measured values of pressure for all Caorso tests. Furthermore, the use of the maximum measured pressure values in the comparison cannot reveal the effect of bubble vertical motion on the measured pressure distribution. Our concerns are that the bubble vertical motion will result in a more severe pressure distribution in the later part of the SRV transient and your model may not yield the correct pressure distribution. Specifically, the cross-correlation coefficient of pressure traces measured at different elevations is less than 1 (about 0.9) which indicates that there may be some effect from bubble motion on the pressure distribution. Accordingly, modify your proposed vertical pressure distribution, as required, to assure conservatism in the design load specifications for SRV transients in the WNP-2 facility.

Response:

a. Proposed Vertical Pressure Distribution

Figure 3.8a of the SRV report may more appropriately be shown as in Figure 022.059-1 such that the measured pressures are normalized to the maximum value of pressure sensor P19 instead of P13, as was used in the original SRV load definition report, since P19 corresponds to the sensor selected in the development of the load definition.

The proposed vertical distribution is compared against the average and maximum values of pressure measured at five locations during six representative tests. Tests 4402, 2325, 2324, 2305, 2202, and 1104 were chosen as representative since these tests measured some of the highest pressures at P19.

This alternate figure shows the proposed vertical distribution as adequate. The pressure distribution which would envelope the maximum values shown in the figure is adequately represented by the proposed pressure distribution.

b. Bubble Vertical Motion

The suggested "bubble vertical motion" will not result in a more severe pressure distribution in the latter part of the transient. It can be shown, using Test 2202 as an example, that the vertical pressure distribution is preserved. Figure 022.059-2 compares the normalized first peak positive and negative maximum pressure amplitudes of pressure sensors located along the vertical face of the containment with the proposed vertical pressure distribution, with satisfactory results. The second positive and negative peak amplitudes as well as the third positive and negative peak amplitudes for the applicable pressure sensors plotted in Figures 022.059-3 and 022.059-4, respectively, again show the distribution as adequate.

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WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	NORMALIZED VERTICAL DISTRIBUTION OF POOL BOUNDARY PRESSURES	FIGURE 022. 059-2

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Q. 022.060

While you discuss both the proposed circumferential and vertical pressure distributions in the report cited above, you do not present a detailed discussion of the radial pressure distribution. Accordingly, indicate how you calculate the radial pressure distribution. State whether you assume the pressure distribution in the radial direction from the reactor pedestal to the containment walls is constant. If not, describe the method you used and demonstrate that your approach is supported by the measured pressure distributions in the Caorso tests.

Response:

Refer to response to Question 022.061.

Q. 022.061

You use the methodology in the DFFR to establish the circumferential pressure distribution for the WNP-2 facility. State whether you use the line-of-sight and square-root-of-the-sum-of-the-squares (SRSS) assumptions of the DFFR in your calculations. If so, provide justification for using these assumptions in the WNP-2 facility. Your justification should be based on the Caorso test results. Indicate to what extent these assumptions affect the WNP-2 load cases. Provide representative figures showing the pressure distributions on the basemat, the pedestal wall, and the containment wall for the various SRV discharge cases considered in WNP-2 plant design assessment.

Response:

The circumferential pressure distribution actually used for the WNP-2 facility is derived from a conservative and more realistic application of the pressure distribution on the suppression pool boundary recommended in DFFR (Reference 1). This conservative application is supported by CAORSO test data as described below.

a. The line-of-sight assumption is not used for WNP-2. To calculate the boundary pressure at a given location resulting from actuation of a single quencher the "straight line" distance is used instead of the "line-of-sight" distance recommended in DFFR:

$$p^{(r)} = \begin{cases} P_B \left(\dfrac{2r_o}{r}\right) & \text{for} \quad \dfrac{2r_o}{r} \leq 1.0 \\ P_B & \text{for} \quad \dfrac{2r_o}{r} > 1.0 \; . \end{cases} \qquad (1)$$

where:

$p^{(r)}$ = attenuated bubble pressure
P_B = bubble pressure
r_o = quencher radius
r = straight line distance from quencher center point to the location of interest.

This results in finite pressure values being calculated over the entire suppression pool boundary, not only over the "viewed" portion of the

boundary, a fact verified by Caorso test data. Indeed, during Caorso Phase II Test 501 X (see Reference 2) quencher "V" was actuated and the available instrumentation, although located in a shaded or "non-viewed" boundary area with respect to the actuated quencher as seen from Figures 4-1, 4-2 and 4-3 of Reference 2, recorded finite boundary pressure values as follows:

Sensor	P_{13}	P_{14}	P_{15}	P_{17}	P_{18}
Recorded pressure value (psi)	0.4	0.4	0.5	0.5	0.7

The bubble pressure value was not recorded during this specific test but can be estimated to lie in the range of values recorded during Phase I Tests 19 and 20, and Phase II Test 22 A01 performed with quencher "U" at similar conditions: single valve first actuations, cold pipe, normal water leg and two 10-inch vacuum breakers (see References 2 and 3). Then, the pressure value recorded at location of sensor P_{17}, located approximately 180° from the actuated quencher "V", is estimated to be in the range of 6.4% to 16.1% of the bubble pressure. This is comparable to the 13% prediction calculated using the "straight line" distance assumption.

b. The SRSS assumption is replaced with the more conservative linear superposition (LS) assumption for the WNP-2 facility. In the case of two quenchers this LS rule becomes:

$$P = P_1 + P_2 \leq P_B \qquad (2)$$

where:

P = total pressure at the location of interest

P_1, P_2 = contributions from quencher #1 and #2, respectively, calculated using eq. (1), above.

Justification for the use of the LS assumption is provided by two Caorso Phase II two-valve tests: Test 24 and Test 25 (see Reference 2). In Table 022.061-1 attached, pressures calculated at different pool boundary locations using the LS rule and the SRSS rule are identified. For comparison purposes pressures recorded at the same locations during the two tests are also listed.

From examination of the data it is concluded that the LS rule is adequate and, as expected, more conservative than the SRSS rule.

Figures 022.061-1 and 022.061-2 illustrate the pressure distributions along the suppression pool floor and on the pedestal and containment walls for two SRV discharge cases, the all-valves case and single outer valve case. These two cases were determined to be the governing cases for the WNP-2 plant design, although the plant was also assessed for a single inner valve discharge case, a two valve discharge case and for the ADS case.

Figure 022.061-3 illustrates the circumferential distribution of pressure loading for the single outer valve case. It should be noted that there is no pressure variation in the circumferential direction for the all valves discharge case.

TABLE 022.061-1

COMPARISON OF WETWELL BOUNDARY PRESSURE AMPLITUDES PREDICTED BY THE LINEAR SUPERPOSITION (LS) METHOD AND THE SRSS METHOD (DFFR, REV. 3) WITH PRESSURES RECORDED DURING CAORSO TWO-VALVE TESTS 24 & 25

SENSOR	PRESSURE AMPLITUDE (PSI)			RATIO	
	CALCULATED BY		MEASURED	L.S. / MEASURED	SRSS / MEASURED
	L.S. METHOD	SRSS METHOD			
TEST 24					
13	5.2	5.2	5.2	1.00	1.00
18[2]	4.7	3.7	3.3	1.42	1.13
23	5.2	5.2	3.6	1.44	1.44
32[2]	4.9	3.5	2.0	2.45	1.75
35	5.2	5.2	4.2	1.24	1.24
36	5.2	5.2	3.1	1.68	1.68
37[1]	5.2	5.2	2.8	1.86	1.86
50	5.2	4.5	3.6	1.44	1.25
51	5.1	3.6	2.1	2.43	1.71
AVERAGE RATIOS				1.66	1.45
TEST 25					
13	3.7	3.7	3.4	1.09	1.09
18[2]	3.6	2.7	2.8	1.29	0.96
23	3.7	3.7	3.3	1.12	1.12
32[2]	3.7	2.7	2.7	1.37	1.00
35	3.7	3.7	3.7	1.00	1.00
36	3.7	3.7	3.2	1.16	1.16
37[1]	3.7	3.7	3.1	1.19	1.19
50	3.7	3.4	3.8[3]	0.97	0.89
51	3.7	2.8	3.0	1.23	0.93
AVERAGE RATIOS				1.16	1.08

For Test 24: $(P_B)_A$ = 5.2 psi, $(P_B)_F$ = 2.7 psi

For Test 25: $(P_B)_A$ = 3.7 psi, $(P_B)_F$ = 2.3 psi

P_B = bubble pressure

$(P_B)_A$ = bubble pressure @ quencher A (recorded by sensor P19)

$(P_B)_F$ = bubble pressure @ quencher F (recorded by sensor P25)

1 Exceeds 2 r_o slightly
2 At quencher elevation
3 Exceeds P_B

<u>References</u>:

1. "Mark II Containment Dynamic Forcing Functions Information Report (DFFR)," NEDO-21061, Rev. 3, dated June 1978.

2. "Mark II Containment Supporting Program. CAORSO Safety Relief Valve Discharge Tests. Phase II Test Report," NEDE-24757-P, dated May 1980, General Electric Company Proprietary.

3. "CAORSO SRV Discharge Tests. Phase I Test Report," NEDE-25100-P, dated May 1979, General Electric Company Proprietary.

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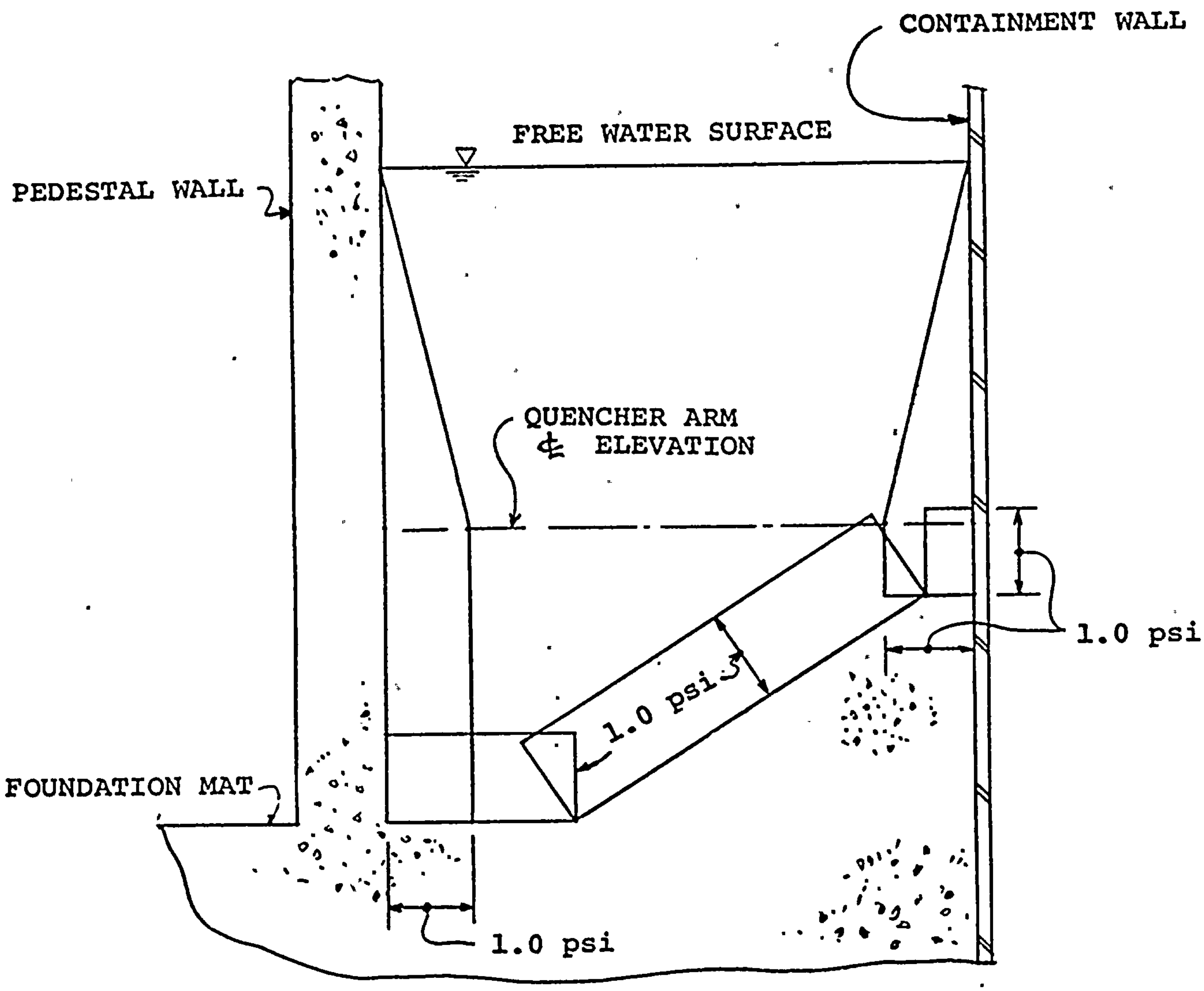

NOTE: FOR ALL VALVES CASE CIRCUMFERENTIAL DISTRIBUTION IS CONSTANT

FIGURE Q 022.061-1

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	ALL VALVES CASE RADIAL PRESSURE DISTRIBUTION CORRESPONDING TO $\theta = 0^{\circ}$

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SINGLE OUTER VALVE

PRESSURE DISTRIBUTION AT 0°

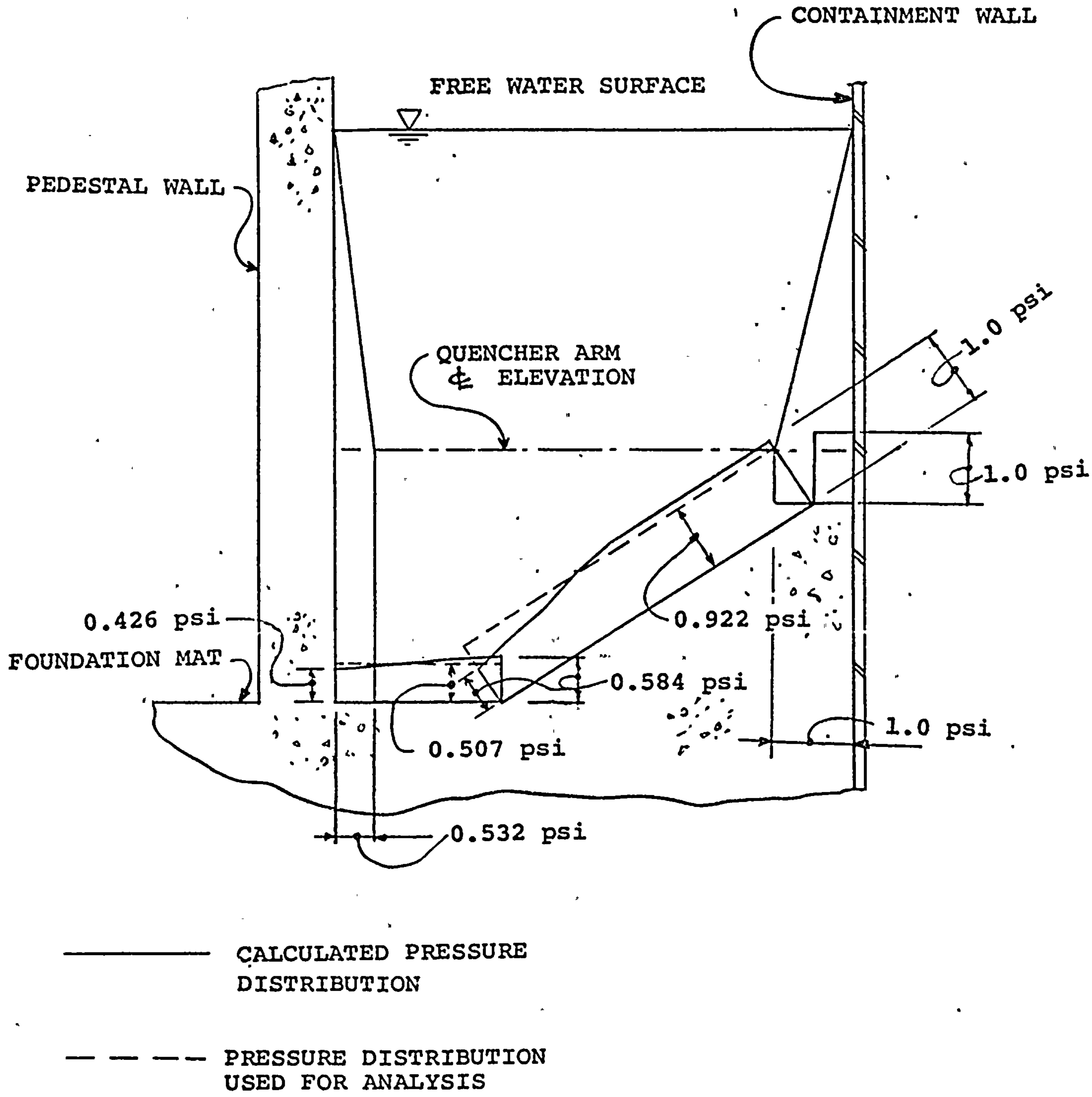

FIGURE Q 022.061-2

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	SINGLE OUTER VALVE CASE RADIAL PRESSURE DISTRIBUTION CORRESPONDING TO $\theta = 0°$

AMENDMENT NO. 20
October 1981

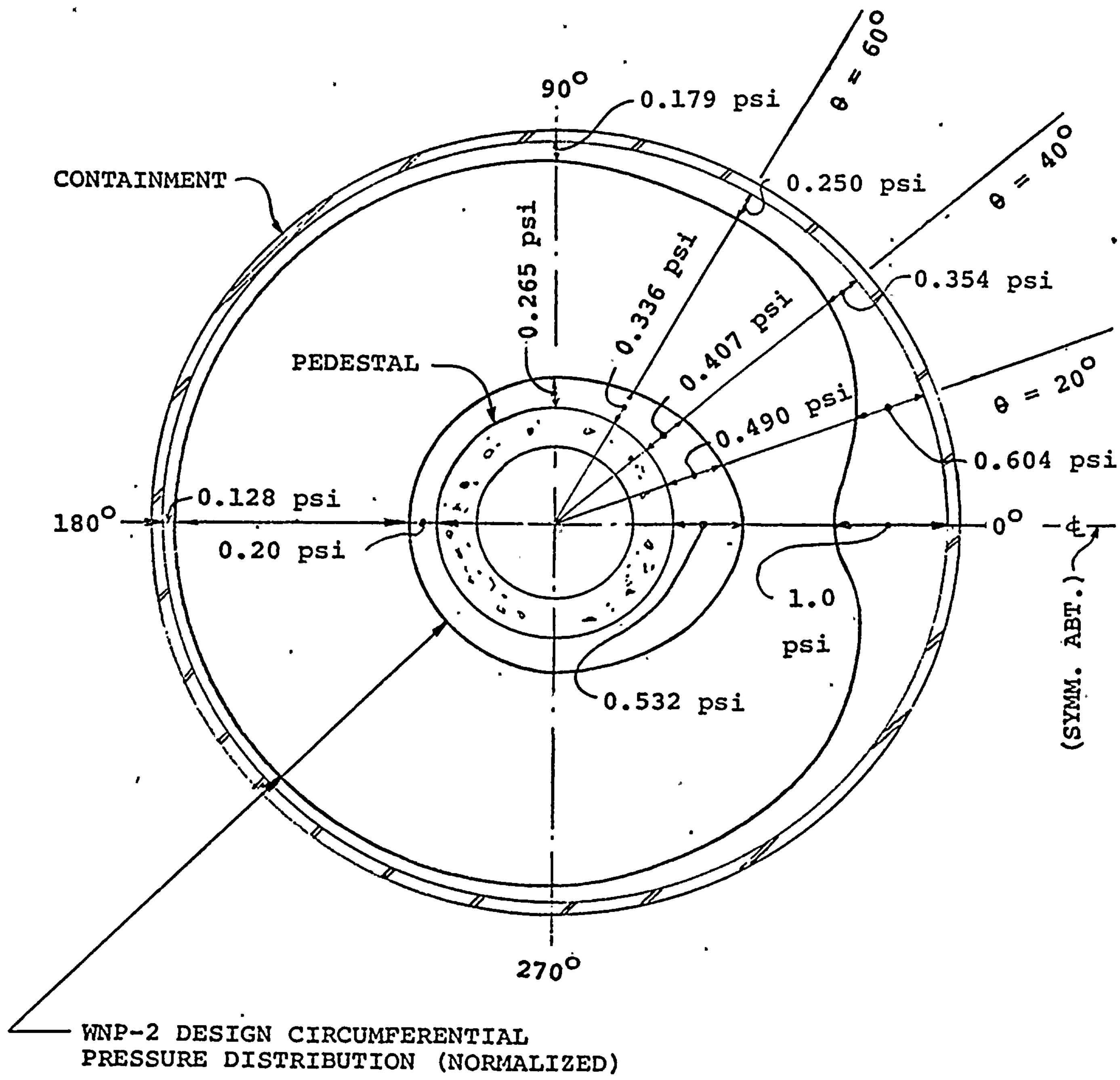

FIGURE Q 022.061-3

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	SINGLE OUTER VALVE CASE CIRCUMFERENTIAL PRESSURE DISTRIBUTION

BLANK

Q. 022.062

Indicate what two valves are selected for the two-valve discharge case. State whether the two quenchers selected are in the inner or the outer circle. Provide justification for the two valves selected and for their location.

Response:

The two valves selected for the two-valve discharge case are identified as 1-B and 3-A; one quencher lies in the inner circle while the other lies in the outer circle. There are a large number of cases to be considered, however, certain DFFR criteria limit this selection. One such criterion specifies that the required combination include one lowset valve and any adjacent valve. This statement limits the number of cases to be considered to seven. Of these seven tests analyzed, the arrangement chosen yields the most critical asymmetric loading condition thereby establishing the basis for the selection. Refer to Figure 022.062-1 for the location of the two selected quenchers.

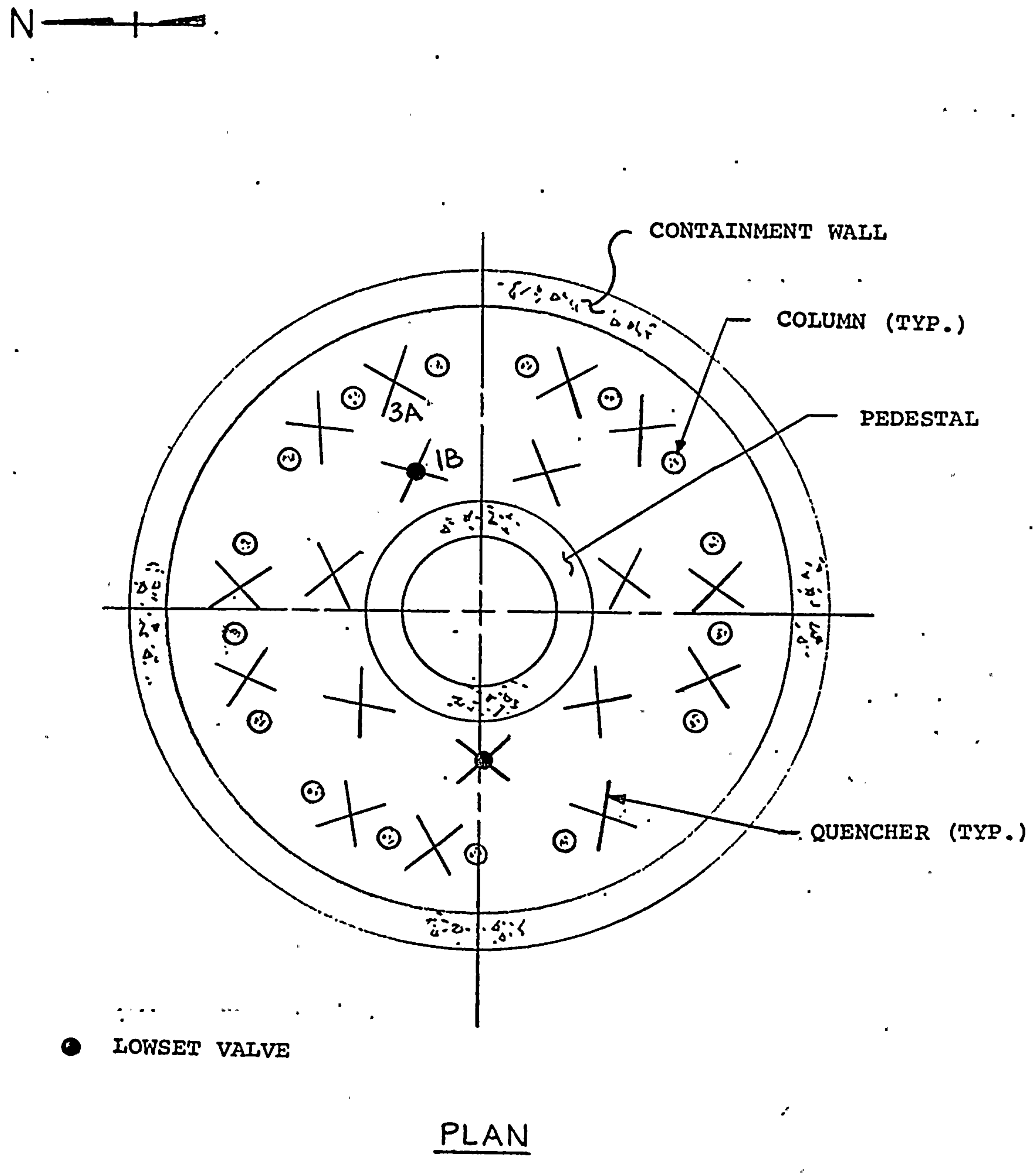

FIGURE Q 022.062-1

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	QUENCHER LOCATION FOR TWO VALVE DISCHARGE CASE

Q. 022.063

In your analyses of SRV transients in the WNP-2 facility, you assume that the pool water is incompressible. Your only justification for this assumption is that the cross-correlation coefficients between pressure time histories measured at different locations are high. Our concern is that this is insufficient justification since the relationship between the cross-correlation coefficient and the time phase shift has not been established; this relationship will influence the effect of compressibility on pressure measurements. You use the incompressible flow assumption in your analyses addressing the fluid-structure interaction (FSI) effect and in the WNP-2 structural analyses. Even though the incompressible flow assumption can be justified for the Caorso plant, it is still questionable whether it holds true for the WNP-2 facility. Specifically, our concern is that the fluid-structure coupling effect may be more significant in the WNP-2 plant which has a steel containment than in the Caorso facility which has a concrete containment. Further, the velocity of sound in water is greatly reduced by the presence of air and steam bubbles in the water; the conditions in the WNP-2 facility may differ to the extent that the amount of air and steam bubbles in the pool water will be significantly different for the two facilities. Accordingly, since your assumption regarding the incompressibility of water in your analyses of SRV transients in the WNP-2 facility is important but not adequately supported, provide additional justification on this matter.

Response:

For any boundary pressure, P, it can be shown that

$$P = P_i + P_a \qquad \text{(See p. 32, SRV report)}$$

where P_i = rigid wall pressure
P_a = interaction pressure due to wall flexibility
= Ma

in which M = hydrodynamic added mass
a = wall acceleration

It was found during the Caorso tests that containment wall acceleration measurements were very small (see Appendix 5-1, SRV report). One may conclude that the interaction pressure, P_a, is to be considered negligible for the Caorso facility so that

$$P \cong P_i$$

The SRV load definition was then based on the rigid wall pressures recorded at the Caorso facility.

In the design assessment phase of the WNP-2 facility, the rigid wall pressure, P_i, was applied at the pool boundary and since the acceleration of the containment wall may no longer be considered negligible (WNP-2, steel containment versus Caorso, concrete containment), the interaction pressure, P_a, is completely accounted for by a set of hydrodynamic added masses using the approach specified in Reference 1.

The high cross-correlation coefficient between pressure-time histories measured at different locations in the Caorso facility is merely a reinforcement of the fluid incompressibility assumption for the Caorso plant.

References:

1. Bedrosian, B., "Analysis of a Mark II Containment Structure for Hydrodynamic Loads in Suppression Pool", Proceedings, Conference on Structural Analysis, Design, and Construction in Nuclear Power Plants, Vol. 2, Porto Alegra, Brazil, April 1978.

Q. 022.064

Provide a list, including appropriate drawings, identifying all piping, equipment, instrumentation and structures in the WNP-2 containment which may be subjected to pool dynamic loads. In addition, provide drawings showing the location of access galleys in the wetwell, the vent vacuum breaker configuration, the wetwell grating, the vent bracing configuration, the vent configuration in the pedestal region of the wetwell and any large horizontal structures in the zone affected by the pool swell phenomenon.

Response:

The "Plant Design Assessment Report for SRV and LOCA Loads," Revision 2, identifies piping, equipment, instrumentation, and structures subjected to pool dynamic loads. Individually, each item is assessed in Chapter 4 of the report. A summary list of items is provided in Table 2.3-1. Drawings are provided in Chapter 2 as Figures 2.1-1 through 2.1-9.

Q. 022.065

Discuss the applicability of the generic supporting programs, tests and analyses (e.g., those relating to fluid-structure interactions, downcomer stiffeners and downcomer diameters) to the design of the WNP-2 facility.

Response:

Please refer to Section 1.1, Role of DFFR/Mark II Program, of the "Plant Design Assessment Report for SRV and LOCA Loads," Revision 2.

Q. 022.066

Provide the time history of plant specific loads and your assessment of responses of plant structures, piping, equipment and components to pool dynamic loads. Identify any significant plant modifications which you made due to considerations of the pool dynamic loads.

Response:

The time history of plant specific loads and the assessment of plant structures, piping, equipment, and components are given in the "Plant Design Assessment Report for SRV and LOCA Loads," Revision 2. Significant plant modifications are given in Section 2.3.1 of the report.

Q. 022.067

Provide the analyses which you performed to determine the post-swell wave load and the seismic slosh load. Discuss your analytical model and the assumptions you made in performing these analyses.

Response:

Please refer to the response to Question 022.020.

Q. 022.068

Provide the type, number and location of the temperature instrumentation which will be installed in the suppression pool for the suppression pool temperature monitoring systems. Discuss the sampling and/or averaging technique that you will use to arrive at a definitive pool temperature. Provide justification for your approach.

Response:

Section 7.6.1.7 has been revised to clarify and more accurately define the WNP-2 suppression pool temperature monitoring, averaging, and annunciation techniques.

Q. 022.069
(RSP)

Based on our review of the information presented in Section 6.2.1.1.5 of the FSAR and your response to Item 022.018 which references your response to Item 031.070, we find that your discussion of steam bypass from the drywell to the wetwell for postulated small steam line breaks, is unacceptable. Specifically, the maximum allowable bypass leakage which you calculate (i.e., $A/\sqrt{K}$ = 0.028 sq. ft.), is not acceptable. Accordingly, we require that you design the WNP-2 containment to have a bypass leakage capability which satisfies the provisions of Appendix I to Section 6.2.1.1.C of the Standard Review Plan (SRP) (i.e., $A/\sqrt{K}$ = 0.05 sq. ft.). Provide the appropriate discussions, justifications and analyses to demonstrate how you comply with the provisions of Appendix I cited above.

Response:

Please refer to revised 6.2.1.1.5.4 and Figure 6.2-17b. Also refer to the revised response to Question 031.070.

Q. 022.070

You state in your response to Item 022.6 that closed systems are not relied upon as barriers to eliminate bypass leakage. However, in your response to Item 022.35, you indicate that the reactor feedwater lines are the only lines for which a water seal is assumed to prevent secondary containment bypass leakage. Accordingly, explain your rationale for eliminating some of the penetrations listed in Table 6.2-16 of the FSAR as potential bypass leakage paths.

Response:

Lines with potential bypass leakage are only those that run through the secondary containment and terminate in the radwaste or turbine buildings (page 6.2-49). All lines which are potential bypass leakage paths are noted in 6.2.3.2 (page 6.2-50).

Q. 022.071

Describe the test which you will perform to verify your assumptions about the amount of inleakage and the drawdown time for reestablishing - 0.25 inches of water gauge in the secondary containment following a postulated loss-of-coolant accident (LOCA).

Response:

The preoperational test for secondary containment leakage will be performed to assure that each train is capable of drawing down the secondary containment to - 0.25 inches of water gauge in 120 seconds and is capable of maintaining this negative pressure at a flow rate not exceeding 2240 CFM.

The leakage test will be performed by using the standby gas treatment filter unit. After system actuation, the air flow through the standby gas treatment filter unit, the secondary containment pressure will be monitored at predetermined time intervals to demonstrate that the system meets the design intent and complies with the SAR and the Technical Specification.

Q. 022.072

DELETED

Q. 022.073

DELETED

Q. 022.074

DELETED

Q. 022.075

DELETED

Q. 022.076

DELETED

Q. 022.077

In Section 6.2.5.2.4 of the FSAR, you state that the WNP-2 containment purge system can be used to perform a controlled purge of the containment atmosphere in the event this is necessary to limit the hydrogen concentration in the containment (e.g., following an accident). We find this approach to be acceptable provided that your purge system is capable of diluting the hydrogen concentration in the containment following a postulated LOCA (i.e., the pressure and temperature in the containment at the time hydrogen purging is required).

Response:

The containment purge system is not required as a backup to the hydrogen recombiners for post-accident hydrogen control inside containment. However, the containment purge system has the capability for a controlled purge of the containment atmosphere to aid in cleanup, per the guidance provided in Section C.4 of Regulatory Guide 1.7.

Sections 6.2.1.1.8.3 and 6.2.5.2.4 have been revised to clarify the post-accident function of the containment purge system.

Q. 022.078

Your response to item 022.048 cited several references and tests conducted to determine the evolution of hydrogen following a postulated LOCA. We are currently undertaking additional effort to better define the various sources of hydrogen, including zinc-rich paints and organic materials. The following equation, which describes the hydrogen generation rates as a function of temperature, is currently used by the staff for its confirmatory analysis.

H_2 (SCF/sq. ft. - hr.) = $4.6 \times 10^5 \exp(-14{,}500/RT)$

where: R (cal/gm K) = 1.986
T = absolute temperature (degrees Kelvin)

We are currently reviewing the information presented in your response to question 022.048. As an acceptable alternative approach to facilitate the staff review, provide a sensitivity study based on the above equation which shows that hydrogen concentration inside the containment will not exceed our acceptance criterion of 4 volume percent. In responding to this question, indicate the time interval following a postulated LOCA at which the hydrogen recombiner should be turned on and the amount of time needed to heat up the recombiner.

Response:

See revised 6.2.5.

Q. 022.079

State the seismic qualification and quality group of the water leg pumps and the associated piping which are used to maintain the water level in the pipes that you identified in your response to Item 022.049, as being filled with water at all times.

Response:

The water leg pumps and associated piping are Seismic Category I and Quality Group B. Table 3.2-1 has been modified to include those items.

Q. 022.080
(RSP)

DELETED

Q. 022.083

The pool dynamic loads resulting from a postulated LOCA which are currently acceptable to the staff are discussed in NUREG-0487, "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria." Specifically, Table IV-1 of NUREG-0487 summarizes these acceptable Mark II pool dynamic loads. To expedite our review of the WNP-2 facility, indicate by referring to Table IV-1, which of our generic criteria will be adopted for the WNP-2 facility. Indicate the alternative criteria that you will use for each item for which an exemption is requested. Provide references which discuss these alternative criteria.

Response:

Please refer to Section 1.1 and Table 1.1-1 of the "Plant Design Assessment Report for SRV and LOCA Loads," Revision 2, for the information requested.

Q. 022.084

Provide the input data for your pool swell model, including all initial and boundary conditions. Demonstrate that the model input represents conservative values of the initial and boundary conditions (i.e., those values which will yield maximum pool swell loads). In the case of input which is calculated (i.e., the drywell pressurization and the vent clearing time), describe and justify your calculational methods.

Response:

The pool swell model, including all initial and boundary conditions, conservative assumptions and calculational methods are discussed in length in Section 3.2 of the "Plant Design Assessment Report for SRV and LOCA loads," Revision 2.

Q. 022.085

Provide in graphic form, the following information:

a. the pool surface velocity versus position; and

b. the maximum pressures of the suppression pool air slug and the wetwell air space.

Response:

a. Plots of the wetwell pool surface motion during a LOCA are presented in Figures 3.2-22 (Velocity vs. Time), 3.2-23 (Acceleration vs. Time), 3.2-24 (Elevation vs. Time), and 3.2-25 (Velocity vs. Elevation) of the "Plant Design Assessment Report for SRV and LOCA Loads," Revision 2.

b. The maximum pressures of the suppression pool air (bubble) slug and the wetwell air space are presented in Figures 3.2-25 and 3.2-26, respectively, of the above referenced document.

Q. 022.086

If your pool swell model is significantly different from the pool swell model previously found acceptable by the staff, compare your calculated drywell pressure response and the enthalpy flux in the downcomer vent with the data obtained from the series of tests conducted in the 4T facility using the 2-1/2-inch and 3-inch venturis.

Response:

The WNP-2 pool swell model conforms to that previously found acceptable by the NRC. Please refer to Table 3.2-2, "Short Term LOCA Hydrodynamic Load Summary Table" in Section 3.2 of the "Plant Design Assessment Report for SRV and LOCA Loads," Revision 2.

Q. 022.087

Provide the information requested in Items 022.084, 022.085, and 022.086, where applicable, for pool swell in the pedestal region.

Response:

Since no structures or downcomers are located in the pedestal region, pool swell effects there are not considered.

Q. 022.088

Your performance tests on a scaled down model of the catalyst bed for the WNP-2 hydrogen recombiner, were conducted in a laboratory test facility in which the gas flow rates and size of the catalyst bed are significantly different than those which will be used in the production model. Indicate the scaling factors used in determining the size of the catalytic bed and the gas flow rates when you established the design of the full-scale recombiner. Provide justification for these scaling factors, including a discussion of the catalyst bed volume, the bed depth, the bed area and any experimental verification of the adequacy of these scaling factors. Additionally, we require you to conduct full-scale tests on a production recombiner unit including the catalyst, to demonstrate that the hydrogen recombiner will perform its intended function in the containment environment which would occur following a postulated LOCA.

Response:

Scaling factors used in determining the size of the catalytic bed and gas flow rates for the full-scale recombiner are discussed in detail in Section 3.1 of APCI-78-6, "Air Products Post-LOCA Recombiner Test Summary", dated June 1978. (APCI-78-6 was submitted to the NRC on the WNP-2 docket by Washington Public Power Supply System letter GO2-78-176, D. L. Renberger to S. A. Varga, "Post-LOCA Hydrogen Recombiner Supplemental Information", dated July 10, 1978.)

As stated in 6.2.5.4, full-scale performance tests of the WNP-2 recombiners were performed. Full scale testing parameters duplicated the feed gas pressure, temperature, flow, steam content, hydrogen and air possible during a postulated LOCA. At no time was the efficiency of hydrogen recombination less than 99%. Detailed information relating to the recombiner tests may also be found in Section 3.3 of APCI-78-6P, the proprietary attachment to APCI-78-6, including effects of halogens on the catalyst in the test recombiner.

Q. 022.089

You conducted catalyst performance tests using a feed gas composition which did not contain steam. Investigate the effect on the various components of the recombiner system and the overall effect on system performance, of having superheated steam in the feed gas.

Response:

As stated in 6.2.5.4, each hydrogen recombiner system has been shop tested. The full-scale performance tests were performed using a feed gas composed of air, hydrogen, and steam at simulated pressure and temperature conditions following a postulated LOCA. At no time during the test was the efficiency of hydrogen recombination less than 99%.

Detailed information relating to the hydrogen recombiner full-scale performance tests is provided in the following Air Products and Chemicals, Inc. reports:

a. APCI-78-6, Air Products Post-LOCA Recombiner Test Summary, dated June 1978. (This report was submitted to the NRC by Washington Public Power Supply System letter number GO2-78-176, D. L. Renberger to S. A. Varga, "Post-LOCA Hydrogen Recombiner Supplemental Information", dated July 10, 1978.)

b. APCI-78-6P, Air Products Post-LOCA Recombiner Test Summary (Proprietary Supplement to APCI-78-6), dated June 1978. (This report was submitted to the NRC by Washington Public Power Supply System letter number GO2-78-200, D. L. Renberger to S. A. Varga, "Post-LOCA Hydrogen Recombiner Supplemental Information", dated August 11, 1978.)

Q. 022.090

In the event of a small steam line break, the potential exists for steam to bypass the suppression pool via the hydrogen control system. Discuss the capability of the hydrogen recombiner system to condense this superheated steam. Discuss whether the after-cooler could be initiated independently from the recombiner system to condense leaking steam. Discuss the design provisions in the WNP-2 facility to eliminate the potential for steam bypass.

Response:

The hydrogen recombiner system containment isolation valves are normally closed and are opened only if the hydrogen concentration inside primary containment requires operation of the recombiner. Therefore, in the event of a small steam line break without hydrogen generation, the isolation valves remain closed and there is no potential for steam bypass through the hydrogen recombiner system.

The hydrogen recombiner system is designed to take suction from the drywell and discharge the processed steam into the suppression chamber. Existing discharge line valves to the drywell and suction line valves from the suppression chamber are key-locked closed and their electrical interlocks with the recombiner are disconnected. The key locks are located on a control room panel for remote manual operation, when and if another mode of operation (based on hydrogen concentration) is required. Refer to 6.2.5.2.3 and Figure 3.2-17. As discussed in the response to NRC Question 022.089, the hydrogen recombiner system is capable of handling a feed gas containing steam. Therefore, in the event of a small steam line break with hydrogen generation and the recombiner system in operation, there is no potential for steam bypass through the hydrogen recombiner system because any steam in the feed gas is condensed in the recombiner's scrubber.

Based on the above discussion and the response to NRC Question 022.089, independent operation of the after-cooler to condense leaking steam is not required.

Q. 022.091

Since the hydrogen recombiner may be required to operate for months following a postulated LOCA, justify the length of time you conducted the performance tests to qualify the production model's capability to perform for extended periods of time.

Response:

All components of the recombiner skid, control panel, and mounted instrumentation were designed for continuous operation when subjected to accident environmental conditions of temperature, pressure, humidity, radiation, and seismic event, following 40 years of periodic cyclic testing under normal environment conditions.

Purchased components had to be tested or analyzed at or above the given service conditions and, in the event of insufficient support data, additional testing was performed. Individual components were then installed in the same configuration as they had been previously qualified, which allowed correlation of test results of the production units.

IEEE reliability qualification is discussed in Section 5.0 of the Air Products and Chemicals, Inc., report APCI-78-6 titled "Air Products Post-LOCA Recombiner Test Summary", dated June 1978. (This report was submitted to the NRC by Washington Public Power Supply System letter number G02-78-176, D. L. Renberger to S. A. Varga, "Post-LOCA Hydrogen Recombiner Supplemental Information", dated July 10, 1978.)

Detailed description of IEEE reliability qualification including actual documentation and test reports is provided in the Air Products and Chemicals, Inc., report titled "IEEE Reliability Qualification Report", Revision A, dated January 7, 1980. (This report was submitted to the NRC by Washington Public Power Supply System letter number G02-80-201, D. L. Renberger to B. J. Youngblood, "Post-LOCA Hydrogen Recombiner Supplemental Information", dated September 16, 1980.)

In addition to the reliability qualification work already performed, documented, and submitted to the NRC, the hydrogen recombiner system has been evaluated per NUREG-0588 criteria and selected components within the system audited by the NRC.

Q. 022.092

The staff guidance in Regulatory Guide 1.7 indicates that equipment for measuring and sampling containment atmosphere should be designed to appropriate engineered safety feature criteria; i.e., Seismic Category I and Quality Group B. Indicate whether the measuring and sampling equipment in the WNP-2 facility conforms to this guidance. Indicate whether the hydrogen analyser is part of the recombiner package and state whether the analyzer catalyst is the same as the recombiner catalyst.

Response:

The WNP-2 containment hydrogen concentration and oxygen concentration are continuously monitored during normal operation and following a postulated LOCA by analyzers that meet the staff guidelines in Regulatory Guide 1.7. Additionally, the hydrogen/oxygen analyzers and associated indications comply with requirements of NUREG-9737. See FSAR Appendix B, Item II.F.1.

The hydrogen/oxygen analyzers are not part of the recombiner package and operate independently from the recombiners. The analyzers are of the thermal conductivity type and do not utilize a catalyst.

Q. 022.093

Provide the results of the testing program for the blower, the preheater, the after-cooler, the water jet eductor and the separator which demonstrate that these components will perform their intended functions in the containment environment which would occur following a postulated LOCA. In your response, include a table of the test parameters and the range over which these parameters were varied. Our position is that your test program should consider the effects of the following variables:

a. irradiation of all components including electrical equipment;

b. seismic conditions;

c. thermal cycling of the equipment and catalyst bed;

d. the temperature of the components and the effluent gas;

e. the air flow rate:

f. the inlet hydrogen concentration;

g. the fission products and their potential for leakage;

h. the steam content.

Response:

The WNP-2 hydrogen recombiner system does not utilize a water jet eductor. However, a comprehensive testing program was performed for the system components as well as a full scale test of the integrated recombiner system under simulated environmental conditions that would occur following a postulated LOCA.

Results of the testing program are provided in the referenced Air Products and Chemicals, Incorporated reports:

a. Radiation

IEEE Reliability Qualification Report, Revision A, dated January 7, 1980. (This report was submitted to NRC by Washington Public Power Supply System letter number G02-80-201, D. L. Renberger

to B. J. Youngblood, "Post-LOCA Hydrogen Recombiner Supplemental Information," dated September 16, 1980.)

b. Seismic Conditions

IEEE Reliability Qualification Report, Revision A.

Dynamic Testing Report for Hydrogen Recombiner System dated January 3, 1980. (This report was submitted to the NRC by Washington Public Power Supply System letter number GO2-80-201, D. L. Renberger to B. J. Youngblood, "Post-LOCA Hydrogen Recombiner Supplemental Information," dated September 16, 1980.)

APCI-78-6, Air Products Post-LOCA Recombiner Test Summary," dated June 1978. (This report was submitted to the NRC by Washington Public Power Supply System letter number GO2-78-176, D. L. Renberger to B. J. Youngblood, "Post-LOCA Hydrogen Recombiner Supplemental Information," dated September 16, 1980.)

Addenda No. 1 to Air Products Post-LOCA Recombiner Test Summary Report No. APCI-78-6P, dated September 19, 1978. (This report was submitted to the NRC by Washington Public Power Supply System letter number GO2-78-223, D. L. Renberger to S. A. Varga, "Post-LOCA Hydrogen Recombiner Supplemental Information," dated September 18, 1978.)

c. Thermal Cycling

Thermal Cycle Test Performance for the Hydrogen Recombiner System, dated December 14, 1978. (This report was submitted to the NRC by Washington Public Power Supply System letter number GO2-80-201, D. L. Renberger to B. J. Youngblood, "Post-LOCA Hydrogen Recombiner Supplemental Information," dated September 16, 1980.)

d. Temperature

IEEE Reliability Qualification Report, Revision A.

e. Air Flow Rate

APCI-78-6, Air Products Post-LOCA Recombiner Test Summary.

f. -

g. Fission Products and Potential Leakage.

APCI-78-6P, Air Products Post-LOCA Recombiner Test Summary (Proprietary Supplement to APCI-78-6), dated June 1978. (This report was submitted to the NRC by Washington Public Power Supply System letter number GO2-78-200, D. L. Renberger to S. A. Varga, "Post-LOCA Hydrogen Recombiner Supplemental Information," dated August 11, 1978.)

All pressure containing equipment including piping between components is considered an extension of primary containment and is subject to appropriate leakage rate testing.

h. Steam Content

APCI-78-6, Air Products Post-LOCA Recombiner Test Summary.

Full scale performance tests of the actual WNP-2 recombiner systems, each consisting of a control panel and skid-mounted process component package, demonstrated a hydrogen recombination efficiency of greater than 99% in all simulated post-LOCA conditions. A table of the test parameters and the range over which these parameters were varied is provided in APCI-78-6, Table 2-1. A summary of performance test objectives can be found in Table 2-3 of the same report.

Q. 022.094

Clearly identify the interfaces between the hydrogen recombiner and the plant design, including a discussion of:

a. whether the recombiner can be operated from the reactor control room;

b. the instrumentation which will be available to the control room operator to permit the operator to monitor the recombiner performance; and

c. any special equipment or power supply needed for the operation of the recombiner.

Response:
Interfaces between the hydrogen recombiner and the plant design are identified on Figure 3.2-17 and discussed in detail in 6.2.5. Specifically:

a. As stated in 6.2.5.1.k:

"The system is designed to operate remotely from the main control room which includes monitoring of combustible gas concentration. The presence of personnel in the vicinity of the operating hydrogen recombiner units is not required."

b. All of the functions necessary to monitor the recombiner performance are located in the main control room and include:

- Process gas flow rate;
- Scrubber water flow rate;
- System pressure at the blower;
- Temperatures at the blower exit, preheater, and recombiner catalytic bed;
- Recycle flow rate.

Primary containment hydrogen and oxygen concentrations are monitored by each of two redundant analyzers that draw samples of the containment atmosphere from both the drywell and wetwell The analyzers are discussed in detail in 7.5.1.5.

c. The recombiners are supplied with redundant Class 1E power. The safety grade cooling water supply, standby service water, is placed into operation by the same signals which start up the ECCS.

Q. 022.095

You state in your proposal for the hydrogen recombiner that the recombiners will be remotely operated. However, you do not discuss the need for gaining access to the combustible gas control equipment area following a postulated LOCA. Discuss the necessity and/or requirements for such access and your proposed criteria for potential radiation exposure to operating personnel. Indicate how you considered these requirements and criteria when you selected suitable locations for the combustible gas control equipment.

Response:

As stated in 6.2.5.1.k, the hydrogen recombiner system is designed to operate remotely from the main control room which includes monitoring of combustible gas concentration. The presence of personnel in the vicinity of the operating hydrogen recombiner units is not required.

Q. 022.096

You state on page 14 of your report, APCI-78-6P, that you calculated an average air velocity of about 3 feet/sec. Provide the analysis, including your assumptions, which you used to calculate that velocity. You also state that the WNP-2 scrubber will operate in a pressure range of 32 to 14 psia and a temperature range of 100 degree F to 220 degree F. Discuss the effect on the scrubber efficiency and the amount of water entrainment if the hydrogen concentration inside containment necessitates operating the hydrogen recombiner at pressures and temperatures outside the ranges cited above (refer to Item 021.004).

Response:

In the discussion on the scrubber in APCI Report No. APCI-78-6P, page 14, APCI stated, "The calculated WNP-2 velocity averages about 3.0 feet per second..." This is the approximate vapor velocity at the bottom of the packed section, based on assumed containment composition, temperature, and pressure. The cross-sectional flow area is based on 14.25-inch inside diameter and a bed porosity of 94% (for 1" Norton #24 Gauge Pall rings). At conditions soon after LOCA (T=215°F, P=31.0 psia, Mol. Wt=23.18), the gas velocity is 2.92 feet per second; this is approximately test condition I, refer to Table 2-1 of APCI-78-6, "Air Products Post-LOCA Recombiner Test Summary," dated June 1978. At conditions long after LOCA (T=100°F, P=14.7 psia, Mol. Wt. = 27.20), the velocity is 2.65 feet per second; this is approximately test condition V, refer to Table 2-1 of APCI-78-6.

At the top of the packed section and at the demister, the velocity is 80 to 100 percent of the velocity at the bottom of the packed section, depending on the extent of cooling of the vapor flow through the scrubber. The effect is more pronounced early in the LOCA. However, the cooling effect is offset by the higher actual containment feed gas flow which was, as tested, approximately ten (10) percent higher than the design flow rates required for each test condition. Hence, it is more accurate to state that the range of velocities is 2.7 to 3.2 feet per second.

The effects on scrubber efficiency of operating at high temperature and pressure are discussed in "Addenda Number One of Air Products Post-LOCA Recombiner Test Summary Report No. APCI-78-6P," Questions 18 and 20.

Q. 022.097

You state on page 68 of your report, APCI-78-6P, that a hydrogen concentration level of 2.5 percent is representative of the hydrogen containment atmosphere. We find that a much higher hydrogen concentration could exist in the containment prior to initiating operation of the recombiner. Accordingly, discuss the applicability of the test you performed using a hydrogen concentration of 2.5 percent in light of the maximum anticipated hydrogen concentration inside containment.

Response:

Section 3.3.4.d of APCI-78-6P describes a unique test that demonstrates the effects of feed gas preheat temperature and hydrogen concentration on catalyst activity in the presence of extremely severe halogen concentration. The hydrogen removal efficiency was recorded over a preheat temperature range of 450°F - 560°F with a feed gas hydrogen concentration of 1.1%. The feed gas hydrogen concentration was then increased to 2.5% and the hydrogen removal efficiency was tested over the same preheat temperature range.

The results of this test are shown in Figure 3-15 of APCI-78-6P and confirm that a 550°F design preheat temperature is conservative even with massive influent iodine concentrations.

APCI-78-6P does not state that a hydrogen concentration level of 2.5% is representative of the hydrogen containment atmosphere. Rather, in the context of comparing 1.1% versus 2.5% hydrogen concentration levels over the same preheat temperature range, it is stated that a hydrogen concentration of 2.5% is more representative of a containment atmosphere with hydrogen accumulation.

Full scale performance tests with up to 4% hydrogen in the feed gas successfully demonstrated a hydrogen removal efficiency of greater than 99%. Details of the full scale performance test may be found in Air Products and Chemicals, Incorporated report APCI-78-6, "Air Products Post-LOCA Recombiner Test Summary," dated June 1978. (This report was submitted to the NRC by Washington Public Power Supply System letter number GO2-78-176, D. L. Renberger to S. A. Varga, "Post-LOCA Hydrogen Recombiner Supplemental Information," dated July 10, 1978.)

Q. 022.098
(6.2.5)

In Table A of your report, APCI-78-7P, you present an analysis of a postulated failure of the containment atmosphere control system which shows that operator action is required to isolate the affected system and to initiate the standby system. Indicate the time interval you assume it would take the operator to complete this action and provide justification for this amount of time. Specify the delay time required for the hydrogen recombiner to reach full efficiency (i.e., that time at which the catalyst bed is preheated to the specified operating temperature). Discuss the effect of this latter delay time on the hydrogen concentration inside containment.

Response:

Each recombiner system consisting of a skid and control panel is independent of the other and the feed to each system is also independent, thus allowing one or both systems to be operated.

The recombiner systems have two manual startup steps for post-LOCA operation. First the systems will go through a warmup of instrumentation and then automatically proceed through preheat of the recombiner to 550°F recycling cover gas. After preheat is complete, a light indicates to the operator that the systems are ready to accept containment gas. It is at this time (approximately 30 minutes after initial startup) that at least one system will receive containment gas. In the event that the first system fails to operate, the second system will be immediately placed into operation. In the event that the first system operates as intended and recombination is verified, the second system could be shut down.

Since the containment will he inerted, oxygen concentration is considered to be of more significance than hydrogen concentration. A description of both hydrogen and oxygen generation in post-LOCA conditions and a discussion of recombiner performance, including startup sequence, may be found in 6.2.5.

Q. 022.099

You state on page 6 of your report, APCI-78-7P, that the recycle flow ratio which you selected is based on the containment pressure to achieve the most efficient system operation. Explain how this approach maximizes the system efficiency. Revise Figure 2.1 of the cited report to include higher containment pressures up to the containment design pressure (refer to Item 022.044). In addition, indicate: (a) the information available to the control room operator to initiate the recycling system; (b) the design criteria for the recycling system; and (c) the type of interlock chosen to preclude inadvertent opening of the recycling system.

Response:

Note: The referenced report should read "Addenda No. 1 to APCI-78-6P" rather than "APCI-78-7P."

In a noninerted containment, the recycle system ratio is set to achieve the most efficient recombiner system operation based upon containment withdrawal rate. Since recombiner flow through the blower on the skid is the summation of containment and recycle flow, reduction of recycle flow will increase containment withdrawal flow. To assist the operator in selecting the proper recycle flow, APCI has generated a curve in the Technical Manual (Figure 2-1) which presents recycle ratio versus containment pressure.

However, since the WNP-2 containment will be inerted, a fixed recycle ratio of 60% was established after completion of the post-LOCA hydrogen/oxygen generation study. Additionally, the controlling document concerning operation of the system is the Operations and Maintenance (O&M) Manual. Figure 2.1 "Recycle Setpoint" is the same as 2-1 in APCI-76-6P. The O&M manual for the hydrogen-oxygen recombiners has been revised to reflect operation of the system in an inerted atmosphere. In the revision, Figure 2.1 has been deleted since the recycle ratio is to be set, and left, at 60%. This is consistent with the discussion of the recycle ratio included in FSAR, Section 6.2.5.

The recycling system does not require operator action for initiation. However, control room instrumentation is available which will provide temperature, pressure, and flow rate information for the system.

The recycle system was designed to provide a means of preheating the recombiner bed prior to introduction of containment gas and to effectively reduce the hydrogen or oxygen concentration, as required, being introduced into the recombiner. This reduction in hydrogen or oxygen is required for levels of 4% for hydrogen or 2% for oxygen to ensure that the recombiner vessel design temperature is not exceeded.

The recycle valve is interlocked with the system feed gas and return gas valves and the scrubber water valve. During warm-up and preheat, the recycle valve is wide open allowing full recycle, and the feed, return and scrubber water valves are closed. Upon completion of preheat, the operator will manually switch the system to the containment feed phase, and this will open the feed, return and scrubber water valves and allow the recycle valve to go to its preset opening. In case of system shutdown, either operator initiated or emergency, the recycle valve will go to its full open position.

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Q. 022.100

You state on page 12 of your report, APCI-78-7P, that the combustible gas control system will perform its intended function at containment pressures up to 26.5 psig even though the system is designed to operate at 16 psig (i.e., the conditions following a postulated LOCA). Provide either your analysis or experimental data which substantiates this statement. Also, discuss the effect on the system performance, of initiating the combustible gas control system immediately following a postulated LOCA. In your response, consider both an inadvertent start and initiation due to a high hydrogen concentration in containment. Indicate the type of interlock chosen to preclude inadvertent operation of the commbustible gas control system. Specify the design conditions of the different components of the combustible gas control system and indicate the effect of high pressure (i.e., the containment design pressure) on the components of the system.

Response:

Note: The referenced report should read "Addenda No. 1 to APCI-78-6P" rather than "APCI-78-7P".

The result presented by Air Products and Chemicals, Inc. (APCI) in "Addenda Number One to APCI Report APCI-78-6P," Question 18, concerning maximum containment pressure of 26.5 psig, was accomplished using a proprietary process fluid program. The parameters assumed and the basic equations utilized by the computer are presented in the addenda, along with the results for containment pressure, temperature, horsepower, flow rate, and preheater electrical load. Based upon no noticeable reduction for recombiner catalyst efficiency during the full scale performance tests at 0 psig to 18 psig, APCI forecasts no catalyst efficiency reduction at 26.5 psig.

The recombiner system does not have a direct high pressure shutdown in case a high pressure feed is introduced, but would safeguard itself with its integral safety relief, high temperature shutdown devices, and the power supply motor overload protection. High feed pressure will result in high blower discharge pressure until the relief device set at 45 psig is actuated. High feed presure also increases fluid density through the blower and will eventually cause a shutdown of the blower motor. In the event that there is high hydrogen concentration in the high pressure feed, excessive temperatures may result in the recombiner or cooler exchanger. The temperature increases would eventually cause system shutdown by the high temperature switches.

Inadvertent operation of the recombiner system under high pressure conditions is prevented by the two-step manual startup procedure. After initial system start there is a 30-minute instrument warmup and preheat-recycle period which must be followed by a second manual start in order to introduce containment feed gas and commence recombination. Therefore, if the combustible gas control system was initiated immediately following a postulated LOCA, it would be at least 30 minutes before the system would be open to containment. During this time period, the postulated high pressure peaks inside containment, which exceed the maximum expected containment pressure of 18 psig during recombiner system operation, will have passed (see Figures 6.2-2, 6.2-6, and 6.2-11).

The design conditions of the different components of the combustible gas control system are provied in APCI-78-6P. In all cases, component design pressures are equal to or greater than the containment design pressure of 45 psig.

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Q. 022.101

In reviewing your report, APCI-73-10, we find that only iodine and methyl iodide were used in the tests conducted to determine the effect of potential poisonous materials on the catalyst. Other materials were not tested because of similar work done by Southern Nuclear Engineering (SNE) and reported in SNE-100. With regard to these tests:

a. Justify why the noble gases and their decay products were not tested, since they will come in contact with the catalyst.

b. Justify why solvents, such as potassium hydroxide and sodium peroxide which dissolve platinum compounds, were not tested.

c. The tests conducted by SNE used a flow velocity which is significantly different than the design flow velocity in the WNP-2 hydrogen recombiner (i.e., the Air Products model). Justify the applicability of the SNE tests to the Air Products hydrogen recombiner.

d. In the SNE tests which were of short duration, various materials which poison the catalyst were found to reduce the efficiency of the catalyst by zero to 17 percent. Discuss the possibility that this performance degradation will increase with time.

e. It is reported that various poisonous materials have only a slight effect on the efficiency of the catalyst. Discuss the cumulative effect on the efficiency of the catalyst, of all the poisons tested.

f. The argument is made in concluding that various poisons will have only a slight effect on the efficiency of the catalyst, that tests were conducted using poison concentrations well in excess of those predicted in containment after a LOCA. However, tests conducted with methyl iodide do not support this argument. In the methyl iodide tests, the poisoning effect did not change as the concentration was reduced. Discuss the possibility of this same effect occurring with other poisons, including the poison which caused a 17 percent reduction in catalyst efficiency.

g. The miscellaneous halide test conducted by SNE showed that the efficiency of the catalyst could be reduced from 50 to 95 percent.

<u>Response</u>:

The response to NRC Question 022.101 is provided in the Air Products and Chemicals, Incorporated report titled "Attachable Amendment No. 1 - Response to Request for Additional Information - Topical Report - Air Products Catalytic Recombiner System - APCI-74-4," dated October 24, 1974 (reference item No. 3, page 4). This report was submitted to the NRC by an Air Products and Chemicals, Incorporated letter to O. D. Parr, Chief, Light Water Reactors Project Branch 1-3, Division of Reactor Licensing, dated October 25, 1974.

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Q. 022.102

Indicate whether all potential catalyst poisons have been tested and justify your response. Demonstrate that the potential catalyst poisons will not have a detrimental effect on the catalyst.

Response:

The response to NRC Question 022.102 is provided in the Air Products and Chemicals, Incorporated report titled "Attachable Amendment No. 1 - Response to Request for Additional Information - Topical Report - Air Products Catalytic Recombiner System - APCI-74-4," dated October 24, 1974 (reference item No. 4, page 10). This report was submitted to the NRC by an Air Products and Chemicals, Incorporated letter to O. D. Parr, Chief, Light Water Reactors Project Branch 1-3, Division of Reactor Licensing, dated October 25, 1974.

Q. 022.103

The concentrations of iodine and methyl iodide used in the tests reported in APCI-73-10 appear to be quite low when compared to the concentrations expected in containment following a postulated LOCA. State the assumptions which you used to determine the concentrations you used and provide appropriate references. In your response, include a discussion of both radioactive and stable isotopes.

Response:

The response to NRC Question 022.103 is provided in the Air Products and Chemicals, Incorporated report titled "Attachable Amendment No. 1 - Response to Request for Additional Information - Topical Report - Air Products Catalytic Recombiner System - APCI-74-4," dated October 24, 1974 (reference item No. 5, page 12). This report was submitted to the NRC by an Air Products and Chemicals, Incorporated letter to O. D. Parr, Chief, Light Water Reactors Project Branch 1-3, Division of Reactor Licensing, dated October 25, 1974.

Q. 022.104

In your discussion in APCI-73-10 of the effect of particulates on recombiner performance, the only solids you considered were the daughter products of radioactive decay of xenon, krypton, iodine and bromine. Additionally, you indicate that 25 percent of the iodine and bromine was assumed to be released to the containment, in accordance with Regulatory Guides 1.3 and 1.4. However, these regulatory guides are to be used in evaluating the radiological consequences following a postulated LOCA; they are not conservative when evaluating the amount of hydrogen released. Accordingly, analyze the effect of particulates on the performance of the hydrogen recombiner if 100 percent of the noble gases, 50 percent of the halogens and one percent of the solids present in the reactor core are released to the containment, as stated in Regulatory Guide 1.7.

Response:

The response to NRC Question 022.104 is provided in the Air Products and Chemicals, Incorporated report titled "Attachable Amendment No. 1 - Response to Request for Additional Information - Topical Report - Air Products Catalytic Recombiner System - APCI-74-4," dated October 24, 1974 (reference item No. 7, page 14). This report was submitted to the NRC by an Air Products and Chemicals, Incorporated letter to O. D. Parr, Chief, Light Water Reactors Project Branch 1-3, Division of Reactor Licensing, dated October 25, 1974.

Q. 022.105

Provide assurance that the catalyst bed will not deteriorate over the 40-year lifetime of the plant. Consider both chemical deterioration of the catalyst and "packing" of the bed.

Response:

The response to NRC Question 022.105 is provided in the Air Products and Chemicals, Incorporated report titled "Attachable Amendment No. 1 - Response to Request for Additional Information - Topical Report - Air Products Catalytic Recombiner System - APCI-74-4," dated October 24, 1974 (reference item No. 9, page 16). This report was submitted to the NRC by an Air Products and Chemicals, Incorporated letter to O. D. Parr, Chief, Light Water Reactors Project Branch 1-3, Division of Reactor Licensing, dated October 25, 1974.

Q. 022.106

Provide a description of the dynamic testing procedures used in establishing the design of the WNP-2 hydrogen recombiner system to withstand vibratory loads arising from seismic events, postulated accidents and other causes (e.g., pool dynamic loads). Describe the methods and procedures you employed to calculate the frequency spectra and amplitudes at the equipment supports of this system. If your analyses and/or testing procedures do not include evaluation of the equipment in the operating mode, indicate how you will assure that this equipment will function when subjected to the combination of seismic loads, accident loads and other vibratory loads. Criteria acceptable to the staff for a seismic qualification program is contained in Sections 3.9.2 and 3.10 of the Standard Review Plan.

Response:

The WNP-2 recombiner skid and control panel have been dynamically tested on a shaker table. The testing was performed in accordance with the recommended practices of IEEE-344, 1975. The results of the testing verified the recombiner system's ability to withstand vibratory loads arising from seismic events, postulated accidents, and pool dynamic loads. The recombiner system operated satisfactorily before, during, and after the dynamic tests.

A detailed discussion of the recombiner system dynamic test program, including the test procedure and response spectra, is provided in the Air Products and Chemicals, Inc. report titled "Dynamic Testing Report for Hydrogen Recombiner System", dated January 3, 1980. (This report was submitted to the NRC by Washington Public Power Supply System letter number GP2-80-201, D. L. Renberger to B. J. Youngblood, "Post-LOCA Hydrogen Recombiner Supplemental Information", dated September 16, 1980).

Q. 022.107

Provide detailed calculations of the wall pressure amplitude multiplier to account for the difference between WNP-2 design conditions and Caorso test conditions.

Response:

Provided below are detailed calculations of the pressure amplitude multiplier developed to convert from Caorso test conditions to WNP-2 design conditions. Table 6.1 of the SRV report defines key plant parameters for single valve actuations important in defining pressure amplitudes for both Caorso and WNP-2 facilities. Test conditions chosen for Caorso were conditions which were typical of subsequent actuations of a single valve.

WALL PRESSURE AMPLITUDE MULTIPLIER
(DFFR Methodology)

I. WNP-2 pressure prediction: (single valve, subsequent actuation)

$VA = 2.423\ m^3$

$AQ = 6.93\ m^2$

$AW = 419\ m^2$

$WCL = 5.3\ m$

$TW = 200°F$

$VOT = 20\ ms$

$MN = 412$ metric tons/hr.

Ref: (See Table 6.1, SRV Report)

$$VAAQ = \frac{VA}{AQ} = \frac{2.423}{6.93} = 0.3496$$

$$MNAQ = \frac{(MN)^{0.7}}{AQ} = \frac{412^{(0.7)}}{6.93} = 9.766$$

$$AWAQ = \frac{AW}{AQ(NN)}$$ Where NN = number of quenchers

$$= \frac{419}{(6.93)\ (1.0)} = 60.46$$

Since VAAQ > 0.255, VAAQ is redefined as 0.255

Since MNAQ > 6.89,

COF = 0.01
MNQJ = MNAQ = 9.766
MNAQ is redefined as 6.89
MNQ1 = MNAQ = 6.89
MNQ2 = $(MNAQ)^2$ = 47.47

Since AWAQ > 20,

AWAQ = 20

$$C = \frac{5\ (F-32)}{9} = \frac{5\ (200-32)}{9} = 93.33$$

LNTW = ln c = 4.536

WCL2 = $(\text{WCL})^2 = (5.3)^2 = 28.09$

AWQ2 = $(\text{AWAQ})^2 = 400.0$

A1 = (VAAQ - 0.1706) (2.58) = (0.255-0.1706) (2.58)
= 0.2178

A2 = 0.

A3 = (MNQ2-52.7) (0.0089) = (47.47-52.7) (0.0089)
= -0.04655

A4 = (MNQJ-6.89) *COF = (9.766-6.89) (0.01) = 0.02876

A5 = (LNTW-3.83) (0.1377) = (4.536-3.83) (0.1377)
= 0.09722

A6 = (WCL-4.0) (0.206) = (5.3-4.0) (0.206) = 0.2678

A7 = (WCL2-16.0) (0.0176) = (28.09-16.0) (0.0176)
= 0.2128

A8 = (VOT-532.0) (0.000148) = (20-532.0) (0.000148)
= -0.07578

A9 = 0.

A10= 0.

PRD 1 = A1 + A2 - A3 + A4 + A5 + A6 - A7 - A8 - A9 + A10
+0.253

PRD 1 = 0.7741 = AA

For subsequent actuations,

CMSA = 1.744

VVPM = 0.012

PROR = 0.229

CONF = 2.065

VVP1 = 0.006

$$AB = (CMSA)^2\ (VVP1) + \left|(VVPM) + \frac{(PROR)^2(CMSA)^2}{(NN)}\right|(AA)^2$$

$$= (1.744)^2(0.006) +$$

$$\left|(0.012) + \frac{(0.229)^2(1.744)^2}{(1)}\right|(0.7741)^2$$

$$= 0.1210$$

AB is redefined as $\sqrt{AB}$ * (CONF)

$$= \sqrt{0.1210}\ (2.065) = 0.7183$$

$$MPPDV = (CMSA)\ (AA) + (AB)$$
$$= (1.744)\ (0.7741) + (0.7183)$$
$$= 2.068 \text{ bar}$$

The predicted peak positive pressure amplitude for single valve, subsequent actuation is

$$2.068\ \frac{(14.7)}{(0.014)} = \underline{29.98 \text{ psid}}$$

II. Caorso Pressure Prediction:
(Single valve, subsequent actuations)

VA = 1.781 m^3

AQ = 6.93 m^2

AW = 370 m^2

WCL = 5.09 m

TW = 90°F
VOT = 45 ms
MN = 390 metric tons/hr.

} Found typical for single valve, subsequent actuation tests at Caorso.

Ref: (Table 6.1, SRV Report)

$$VAAQ = \frac{VA}{AQ} = \frac{1.781}{6.93} = 0.2570$$

$$MNAQ = \frac{MN}{AQ}^{(0.7)} = \frac{(390)^{0.7}}{6.93} = 9.397$$

$AWAQ = \frac{AW}{AQ(NN)}$ Where NN = Number of quenchers

$= \frac{370}{(6.93)\ (1.0)} = 53.39$

Since VAAQ > 0.255, VAAQ is redefined as 0.255

Since MNAQ > 6.89,

COF = 0.01

MNQJ = MNAQ = 9.397

MNAQ is redefined as 6.89

MNQ1 = MNAQ = 6.89

$MNQ2 = (MNAQ)^2 = 47.47$

Since AWAQ > 20,

AWAQ = 20

$C = \frac{5}{9}\ (F-32) = \frac{5}{9}\ (90-32) = 32.22$

LNTW = ln c = 3.473

$WCL2 = (WCL)^2 = (5.09)^2 = 25.91$

$AWQ2 = (AWAQ)^2 = 400.0$

A1 = (VAAQ-0.1706) (2.58) = 0.2178

A2 = 0.

A3 = (MNQ2-52.7) (0.0089) = -0.04655

A4 = (MNQJ-6.89) (COF) = 0.02507

A5 = (LNTW-3.83) (0.1377) = -0.04916

A6 = (WCL-4.0) (0.206) = 0.2245

A7 = (WCL2-16.0) (0.0176) = 0.1744

A8 = (VOT-532.0) (0.000148) = -0.07208

A9 = 0.

A10= 0.

PRD1 = A1 + A2 - A3 + A4 + A5 + A6 - A7 - A8 - A9 + A10 +0.253

PRD1 = AA = 0.6154

For subsequent actuations,

CMSA = 1.744

VVPM = 0.012

PROR = 0.229

CONF = 2.065

VVP1 = 0.006

$$AB = (CMSA)^2(VVP1) + \left| (VVPM) + \frac{(PROR)^2(CMSA)^2}{(NN)} \right| (AA)^2$$

AB = 0.08320

AB is redefined as $\sqrt{AB}$ * CONF

AB = 0.5956

MPPDV = (CMSA) (AA) + (AB)

= 1.669 bar

The predicted peak positive pressure amplitude for single valve, subsequent actuation at Caorso is:

$$1.669 \ \frac{(14.7)}{(1.014)} = \underline{24.20 \text{ psid}}$$

PRESSURE AMPLITUDE MULTIPLIER

Conversion from Caorso test conditions to WNP-2 design conditions

$$C, = \frac{29.98}{24.20} \approx 1.2$$

REFERENCES:

1. Letter, D. L. Renberger to B. J. Youngblood, "Submittal of SRV Report", dated August 8, 1980, GO2-80-172, transmitting report titled "SRV Loads - Improved Definition and Application Methodology for Mark II Containments".

Q. 022.109

Provide the quencher submergence and SRV line volumes for all WNP-2 discharge lines.

Detailed quencher design and vacuum breaker characteristics are important in the determination of SRV air clearing load. Due to the difference in detailed quencher design and vacuum breaker characteristics between Caorso and WNP-2, we require further justification of the applicability of Caorso data to WNP-2 or require in-plant test.

Response:

Table 022.109-1 provides the quencher submergence and SRV line volumes for all 18 SRV discharge lines at WNP-2. The comparison between the quenchers at Caorso and WNP-2 is discussed in the response to Question 022.053 and the vacuum breaker comparison is addressed in Question 022.054. In both comparisons there appear to be no significant differences that would substantially affect the SRV air clearing load. For the details of the responses refer to the above referenced questions. Based on these comparisons, an in-plant SRV test at WNP-2 does not appear to be required.

TABLE 022.109-1

QUENCHER SUBMERGENCE AND SRV DISCHARGE LINE AIR VOLUMES

Valve No.	Length (ft.) 10"Ø	12"Ø	Total (ft.) (1)	Submergence (ft.) (2)	Air Volume ($ft.^3$) (3)
1A	104.5	31.96	136.48	17.3	65.1
2A	106.81	34.96	141.77	17.3	68.3
3A	109.20	42.97	152.17	17.3	75.2
4A*	127.98	29.95	157.93	17.3	75.4
1B	91.53	30.00	121.53	17.3	57.2
2B	108.11	35.67	143.77	17.3	69.5
3B	131.04	34.96	166.01	17.3	80.4
4B*	118.54	51.19	169.74	17.3	85.6
5B*	109.28	38.67	147.96	17.3	72.2
1C	101.92	30.00	131.92	17.3	62.4
2C	129.31	34.05	163.36	17.3	78.9
3C	141.79	30.67	172.46	17.3	82.8
4C*	136.72	29.96	166.67	17.3	79.7
5C*	126.47	49.04	175.53	17.3	88.1
1D	84.36	43.17	127.53	17.3	62.9
2D	118.57	45.27	163.84	17.3	81.5
3D*	110.44	34.95	145.39	17.3	70.1
4D*	106.38	44.17	150.55	17.3	74.6

*ADS Valves

NOTES: 1. SRV line to the top of quencher

2. High water level (El. 466.40 ft.) to the ℄ of a quencher arm. (Top of quencher to the ℄ of arm = 3.75 ft.)

3. 10" and 12" - Sch. 80

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Q. 022.110

Our evaluation of the Caorso data reveals that higher wall pressure amplitudes are observed for consecutive SRV actuation tests for lines with two 10" vacuum breakers than those with only one vacuum breaker. Since the WNP-2 design utilizes two 10" vacuum breakers on each SRV line, it is our position that pressure amplitude multipliers which will account for this difference should be provided.

Response:

Please refer to the response to Question 022.057.

Q. 022.111

Our evaluation of the Caorso data indicates that higher pressure amplitudes are observed for multiple SRV actuation tests than single SRV first actuation tests. Since WNP-2 specifications are based on single SRV actuation test results, it is our position that a pressure amplitude multiplier for the all-valve case based on the DFFR correlation (assuming WNP-2 surface area) should be used.

Response:

Please refer to the response to Question 022.055.

Q. 022.112

The vertical wall pressure distribution in the WNP-2 specification does not bound Caorso test results. Since the accuracy of sensors used to obtain test data is questionable, it is our position that the staff generic acceptance criteria set forth in NUREG-0487, Supplement 2, Item II.B.4.d should be used.

Response:

As indicated in the response to Question 022.059 and illustrated in Figure 022.059-1, the vertical wall pressure distribution in the WNP-2 specification does bound Caorso test results. Furthermore, the vertical wall pressure distribution in the WNP-2 specification was also verified by TOKAI-2 test results, as shown in Figure 3.8b of the Reference 1 report. Plant assessments are being performed using the vertical wall pressure distribution defined in the SRV report. A major effort would be involved in adopting an alternative vertical wall pressure distribution which does not appear to be warranted based on existing test data.

REFERENCES:

1. Burns and Roe, Inc., "SRV Loads - Improved Definition and Application Methodology to Mark II Containments - Technical Report", dated July 29, 1980 (proprietary), submitted to NRC by WPPSS to NRC letter GO2-80-172, "Submittal of SRV Report", August 8, 1980.

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Q. 022.113

The method used in the calculation of the circumferential pressure distribution in the WNP-2 asymmetric case may not be conservative because of an over-prediction of pressure on the opposite side of the pool of the discharging quencher(s).

It is our recommendation that zero dynamic pressure be specified for the 180° circumference on the opposite side of operating quenchers to assure a maximum overturning moment.

Response:

Please refer to the response to Question 022.061.

Q. 022.114

The use of the DFFR correlation in the calculation of pressure multipliers to account for differences in parameter values between the WNP-2 design condition and Caorso test conditions is not necessarily conservative.

Over-prediction of pressure amplitude corresponding to the Caorso test conditions by the DFFR correlation may lead to under-prediction of the pressure multiplier. Furthermore, despite the overall conservatism in the DFFR correlation, trends with respect to individual parameters may not be conservative, e.g., trend with respect to SRV steam flow.

It is, therefore, our position that trends obtainable from Caorso test results, if more conservative than the DFFR correlation should be used in the pressure multiplier calculations or incorporation of the Caorso data in the DFFR model should be provided for our review.

Response:

Please refer to the response to Question 022.058.

Q. 031.001 (a)

The FSAR contains many conflicting statements and incomprehensible statements which must be resolved prior to the start of our review. For each of the items below, provide a response which is responsive to the NRC staff's need for information satisfying the requirements of the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," (Revision 2).

a. Clarify the discrepancy between the designation of the main generator as "Unit 1" in 1.2.2.7.1 and the plant designation as WNP-2.

RESPONSE:

The text of 1.2.2.7 has been revised to clarify this discrepancy.

Q. 031.001 (b)

Clarify the discrepancy between the reference to "three trip logics" in 3.1.2.3.2.1 and the description of the reactor protection system in 7.2.1.1.3.2 and Figures 7.2-3 and 7.2-11.

Response:

Section 3.1.2.3.2.1, third paragraph which refers to "three trip logics" is in error, and has now been revised.

Amendment 10, revising Chapter 7, replaced 7.2.1.1.3,2 with 7.2.1.1. Figure 7.2-11 has been deleted.

Q. 031.001 (c)

Clarify the discrepancy between the 0.25 to 33 Hz sesimic range given in 3.10.1.2.3.1 and the 5 to 33 Hz values given in 7.3.2.1.2.3.1.

Response:

Some seismic vibration tests on certain equipment were performed starting from 5 Hertz because of test machine limitations. High test table displacements beyond the capacity of the testing machine are required to obtain higher input accelerations (g-levels) at frequencies below 5 Hertz. In all these cases it was shown by calculations or by a resonant frequency search test or based on similarities with previously qualified equipment that no resonant frequencies exist below 5 Hertz, consequently, tests are completed to required levels at 33 Hertz and at all resonant frequencies in between. This would qualify the equipment seismically. Section 7.3.2.1.2.3.1 has been revised to eliminate the discrepancy, and will cross-reference to 3.10 for seismic data.

Amendment 10, revising Chapter 7, replaced 7.3.2.1.2.3.1 with 7.3.2.1.2.a.

Q. 031.001 (d)

Indicate whether the reference to Figure 7.2-4 which is made in 7.2.1.1.3.2, is intended to be Figure 7.2-3.

Response:

Section 7.2.1.1.4.3 has been revised to provide the correct figure references.

Amendment 10, revising Chapter 7, replaced 7.2.1.1.4.3 with 7.2.1.1.b.

Q. 031.001 (e)

Describe, in accordance with the request of 7.3 of the Standard Format, the design features that prevent rapid closure of the recirculation flow control valve during a postulated loss-of-coolant accident thereby permitting the classification of this valve as inactive and excluding it from Class 1E requirements which are referenced in 5.2.2.4.3.2.2.

RESPONSE:

As a result of the installation of the ASD, the flow control valves are no longer needed. Therefore, they have been mechanically disabled in the full open position.

Q. 031.001 (f)

Quantify the accuracy of the temperature detectors, signal processing equipment and bistable devices which are described as "accurate" on Page 5.4-3 of 5.4.1.3 of the FSAR.

RESPONSE:

The system accuracy of the temperature detectors, signal processing equipment and bistable devices is $\pm$ 0.2^{O}F at the 95% (two standard deviation) confidence level. 5.4.1.3 has been revised to incorporate the response to this question.

Q. 031.001 (g)

Clarify the discrepancies between the main steamline isolation valve (MSIV) response time given in 5.4.5.3, 6.3.3.3.1, and 7.3.2.1.2.3.1.5.2.1.1 and the accident analyses of Chapter 15.

Response:

No discrepancies exist between the stated MSIV response times. The main steamline isoation valve closure time is in the range from 3 to 5 seconds. As indicated in 7.3.2.2.2.3.1.1 the minimum MSIV closure time is 3 seconds. Section 5.4.5.3 assumes a maximum MSIV response time of 5.5 seconds. This assumes the maximum closure time of 5 seconds for MSIV plus 0.5 second for instrument response to initiate MSIV closure. The analyses of Chapter 15 assume the worst case of response of 3 seconds for MSIV closure. All closure times given are within the specified 3 to 5 second range of the MSIV.

The MSIV response time for full closure is set prior to plant operation. The minimum set time for closure is 3.0 seconds and the maximum set time is 5.5 seconds (includes as much as 0.5 second for instrument response). The difference in response time reported in the various paragraphs is due to whether the minimum or maximum response time is conservative to the results of the analysis under consideration. Sections 5.4.5.3 and 15.6.4 list the maximum value while 7.3.2.2.2.3.1.1 lists the minimum value. In summary, the conservative response time for MSIV closure was used in each of the identified sections of the FSAR.

Amendment 10, revising Chapter 7, replaced 7.3.2.2.2.3.1.1 with 7.3.2.1.2.a.

Q. 031.001 (h)

Clarify the discrepancy between the description of the MSIV solenoid valves which is given on Page 6.2-55 in 6.2.4.2 and that presented in 7.3.1.1.2.2 and Figure 7.3-19.

Response:

Section 7.3.1.1.2.4 and Figures 7.3-2 and 7.3-4 are correct. MSIV solenoid valves are no longer discussed in 6.2.4.

Amendment 10, revising Chapter 7, replaced 7.3.1.1.2.4 with 7.3.1.1.2.b.

Q. 031.001 (i)

Resolve the discrepancies in 6.2.4.2.1.2 between: (1) the design basis for manual control of the feedwater isolation valve; (2) the use of turbine driven feed pumps which need steam from the main steam system to operate; (3) the generic ATWS study which assumes that motor-driven feed pumps are used and are designed to trip at the start of an accident.

RESPONSE:

1& 2. Short term containment isolation of the feedwater lines is achieved by inboard simple check valves and outboard positive acting check valves. This design accomplishes the intent of isolation (prevents release of inventory from the containment through the feedwater lines), but maintains the option of using the feedwater system as a makeup source even after a containment isolation.

Although the turbine driven feed pumps will become ineffective soon after the main steam lines are isolated, makeup water can be supplied through the feedwater lines with the condensate system pumps if reactor pressure can be reduced to below the discharge pressure of the condensate system. Therefore, even after a containment isolation occurs, it is advisable to keep the option of using the feedwater lines as a makeup path to the reactor.

The motor operated isolation valves are provided to give more positive leakage control on a long term basis. After the short term inventory demands of the reactor have been satisfied, the motor operated isolation valves can be closed by the operator to provide positive isolation of the feedwater lines.

3. No significant inconsistancies are believed to exist between this (or any other) SAR section and the ATWS analyses submitted by General Electric. The generic ATWS studies applicable to the Hanford Unit (Reference 1 and 2) assumed the use of turbine driven feed pumps and simulated the loss of steam to the turbine and

feedwater flow in the most limiting case in which all main steamlines were isolated. In the ATWS situation, the loss of feedwater flow (or limiting of the flow to near zero) causes a decrease in core flow and inlet subcooling which results in a power reduction. This leads to a benefit in mitigating the peak vessel pressure, containment pressure and suppression pool temperature. Therefore, plants with motor driven feed pumps will also be limited to low flow for achieving the same benefit as turbine driven plants.

Since WNP-2 has turbine driven feed pumps, the ATWS studies (Reference 1 and 2) can be applied. The analysis discussed in 15.8 is consistent with this plant response.

See also revised 6.2.4.3.2.1.1.1.

Reference:

1. "Studies of BWR Design for Mitigation of Anticipated Transients Without Scram," NEDO-20626, October 1974.

2. General Electric Response to NRC Status Report, "General Electric ATWS Report," and Appendixes, June and September, 1976 (Proprietary).

Q. 031.001(j)

Provide justification for not seismically qualifying the feedwater and control rod drive excess flow isolation valve actuators.

Response:

Seismic qualification documentation for the Class IE feedwater isolation valve actuators has been assembled as part of an overall qualification actuators review. These actuators are qualified.

The control rod drive excess flow isolation valve has been deleted along with CRD Return Line.

Q. 031.001(k)

Provide justification for not environmentally qualifying the feedwater and control rod drive excess flow isolation valve actuators.

Response:

Environmental qualification documentation for the Class IE feedwater isolation valve actuators has been assembled as part of an overall qualification review. These actuators are qualified.

The control rod drive excess flow isolation valve has been deleted along with the CRD Return Line.

Q. 031.001 (1)

Clarify the discrepancy between 6.2.4.2.2.4 (sic) and 6.2.4.2.2.4 and Figure 7.4-1b which describe the LPCS, LPCI, HPCS, and RCIC pump bypass valves as manually actuated, and Figures 6.3-1a, 6.3-1c, 6.3-4a, and 7.4-2b and 7.3.1.1 which indicate that these valves are automatically controlled.

RESPONSE:

There is no discrepancy observed. 6.2.4.3.2.2.1.1 and 6.2.4.3.2.2.1.2 refer to remote-actuated valves in the bypass lines. These valves, as Figure 7.4-1a shows, are both manually and automatically actuated. On pump high discharge pressure the valves automatically open and on high flow in the discharge line the valves close. Remote manual operation is also possible.

Q. 031.001(m)
(6.2.5)

Clarify the discrepancy between 6.2.4.2.2.7, which discusses an inerting system, and 6.2.5.1.b(c), which indicates that inerting is not necessary, so that the staff can evaluate the effect of the environmental conditions inside of containment on instrumentation and control equipment.

Response:

Since the NRC staff position of April 2, 1981, requiring that "the General Electric pressure suppression containment systems identified by Mark I and Mark II, be inerted", the WNP-2 containment design incorporates a nitrogen inerting system. The Containment Nitrogen Inerting System design is discussed in 6.2.5.7. Section 6.2.5.1 has been revised to reflect operation of the Containment Atmosphere Control System in an inerted atmosphere (6.2.5.1.d). Please note that Section 6.2.4.2.2.7 has been deleted.

Q. 031.001 (n)

Clarify the discrepancy between the single stuck rod criterion stated in 6.3.1.1.3 and the 20 inoperable rod criterion which is a design basis for the rod sequence control system (submitted separately as a generic study).

RESPONSE:

The single stuck rod criterion is no longer discussed in Chapter 6. However, as previously written 6.3.1.1.3, "Reactivity Required for Cold Shutdown", stated:

"The reactor is designed to be in the cold shutdown condition with control rod of highest worth reactivity fully withdrawn and all other control rods fully inserted. (Refer to 4.3.2 for discussion)".

This is not a single stuck rod criterion, it is a definition of required shutdown margin. Per Tech Spec regulation, the operator must demonstrate prior to startup that this margin exists. In case of one or more stuck rods, core conditions have changed and shutdown margin must be demonstrated anew. If the margin test or analysis fails the reactor must be shutdown.

The Rod Sequence Control System has the capacity to allow for up to 20 inoperable rods; the actual built-in maximum is 8 rods.

The requirements to demonstrate the required shutdown margin for the case of one or more stuck rods are now discussed in Chapter 16.

Q. 031.001 (o)

Supplement the FSAR with quantitative information in place of statements such as: (1) "sufficient quantity" (6.3.3.8); (2) "minimum storage" (6.3.3.8); (3) "reserve level" (6.3.4); and (4) "significantly" (7.2.1.1.3.1).

RESPONSE:

1. The minimum volume or "sufficient quantity" of water in the suppression pool is given in Table 6.2-1 as 112,197 ft^3.

2. The "minimum storage" in the condensate storage tank is provided in 9.2.6.3 as 135,000 gallons.

3. The "reserve level" in the condensate storage tank is that level corresponding to the minimum capacity given in 9.2.6.3 as 135,000 gallons.

4. 7.2.1.1.3.1, Initiating Circuits, stated the following:

   "The reactor protection system instrumentation, shown in Figure 7.2-2, is discussed in the following paragraphs."

   The word "significantly" did not appear in the statement and therefore a response cannot be provided. A check of the following paragraphs did not produce the word "significantly" either. Chapter 7 has subsequently been revised and 7.2.1.1.3.1 was eliminated.

See also revised 6.3.4.2.1 and 6.3.3.5.

Q. 031.001 (p)

Clarify the discrepancy between the reference to drywell external relief valves in 6.2.1.1.2.c and the description of the suppression pool vacuum breakers described in 3.8.2.1.3 so that the staff can evaluate the position indication instrumentation.

RESPONSE:

The text of 3.8.2.1.3 and 6.2.1.1.2 have been revised to incorporate the response to this question.

Q. 031.001 (q)

Clarify the discrepancies between 6.3.3.8, which implies reliance of: (1) administrative control; or (2) a 10 minute timer for the containment spray interlock, and 6.5.2.2 and 7.3.1.1.1.4.1.4 which state that the interlock is initiated by the pressure vessel level.

RESPONSE:

The above interlocks are discussed in revised 6.2.2.2, 6.5.2.2 and 7.3.1.1.4. The discrepancies have been eliminated.

Q. 031.001 (r)

The discussion in Chapter 7 regarding compliance with the requirements of 4.20 of IEEE Std. 279-1971 relating to the readout of information, is inadequate. Revise the FSAR to describe the equipment and systems which provide the operator with accurate, complete, and timely information pertinent to the status of the information channel and to generating station safety.

RESPONSE:

The information required to identify compliance with the requirements of 4.20 of IEEE Std. 279-1971, i.e., descriptions of the equipment and systems which provide the operator with accurate, complete and timely information, is in general provided in the sections titled, "Reactor Operator Information," under the appropriate system in the FSAR. The FSAR has been revised to include the appropriate references within the subparagraphs specifically addressed to 4.20. Following is a list of paragraphs in Chapter 7, specificallly related to 4.20 and the paragraphs which are referred to:

Paragraph 7.2.2.1.2.3.1.20 (Reactor Protection System) refers to paragraph 7.2.1.1.6.1.

Paragraph 7.3.2.1.2.3.1.20 (ECCS) refers to paragraphs 7.3.1.1.1.3.11.2 (HPCS), 7.3.1.1.1.4.11.2 (ADS), 7.3.1.1.1.5.11.2 (LPCS), and 7.3.1.1.1.6.11.2 (LPCI).

Paragraph 7.3.2.2.2.3.1.20 (PCRVICS) refers to paragraphs 7.3.1.1.2.4.1.2.9, 7.3.1.1.2.4.1.7.9.2, and 7.3.1.1.2.13.2.

Paragraph 7.3.2.3.2.3.1.20 (MSLIV-LCS).

Paragraph 7.3.2.4.3.1.20 (CSCS) refers to 7.3.1.1.4.12.2.

See 7.3.2.5.20 for SSW system information readout.

See 7.3.2.6.20 for main control room HVAC system information readout.

See 7.3.2.7.20 for reactor building ventilation and pressure control system information readout.

See Chapter 8 for standby power system information readout.

See 7.3.2.9.20 for SGTS information readout.

See 7.3.2.10.20 for CIA system information readout.

See 7.3.2.11.20 for CAC system information readout.

Paragraph 7.4.2.1.2.3.1.20 (RCIC) refers to paragraph 7.4.1.1.5.2.

Paragraph 7.4.2.2.2.3.1.20 (SLCS) refers to paragraph 7.4.1.2.5.2.

Paragraph 7.6.2.8.2 (Recirculation Pump Trip System) refers to paragraph 7.6.1.8.5.2.

Section 7.5.2, Safety-Related Display System Analysis, contains further discussion of compliance with 4.20 of IEEE Standard 279-1971.

Amendment 10, revising Chapter 7, replaced sections describing "Reactor Operator Information" with a description of system operation, stating the applicable figures and tables containing the information. For example, 7.2.1.1.b, RPS Operation, states the figure and table numbers for mechanical equipment and information displays, component control logic, instrument specifications, channel and logic arrangement, actuator and logic arrangement, trip system logic and sensor input arrangements. Compliance to 4.20 of IEEE Standard 279-1971 4.20 is discussed in 7.3.2.1.2(2). See the section of Chapter 7 describing a specific system for the information on that system. For example, ECCS is described in 7.3.1.1.1 and PCRVIS in 7.3.1.1.2.

Q. 031.001 (s)

Clarify Table 7.1-1, Figure 7.3-8e, and 7.6.1.10 and 7.7.1.7 since it is the staff's understanding that the prompt relief trip (PRT) has been deleted from the GE design for the BWR-5.

RESPONSE:

References to the prompt relief trip (PRT) and the relief valve augmented bypass (REVAB) have been deleted from Chapter 7 of the FSAR.

Q. 031.001 (t)

Indicate which penetration option of Table 7.1-3 is applicable to the WNP-2 design.

RESPONSE:

Table 7.1-11 has been revised to incorporate the response to this question.

Q. 031.001 (u)

Clarify the references in 7.2.1.1.4 to Table 3.11-1 for the reactor and control building environments.

Response:

Section 7.2.1.1.5 of revised text states that the environmental conditions for the drywell, the containment, and the turbine building are given in Tables 3.11-1, 3.11-2, and 3.11-3.

Tables 3.11-1, 3.11-2 and Ref 3.11-2 have been revised to provide the referenced information.

Amendment 10, revising Chapter 7, replaced 7.2.1.1.4 with 7.2.1.2.f.

Q. 031.001 (v)

Clarify 7.3.1.1.2.3 to clearly state where the pressure, temperature, and water level sensors and racks are located.

Response:

The text of 7.3.1.1.2.3 and Table 7.3-46 has been revised to incorporate the response to this question.

Amendment 10, revising Chapter 7, deleted Table 7.3-46.

Q. 031.001 (w)

Clarify the discrepancy between the discussion of compliance with the requirements of 4.10 of IEEE Std. 279-1971 in 7.3.2.1.2.3.1.5.2.1.10 and the discussion of conformance with the staff positions in Regulatory Guide 1.22 which follows it.

Response:

Even though the main steamline high temperature sensors are inaccessible during plant operation, they can be tested while the plant is operating by cross comparison between channels.

Section 7.3.2.2.2.1.2 reads in part as follows:

> The main steamline isolation logic, and sensor devices (except the MSL high temperature sensors) may be tested from the sensor device to one of the two solenoids. Both solenoids must be de-energized to verify that there are no obstructions to the valve stem at full power. A reduction in power is necessary to avoid reactor scram before performing a valve closure using two, fast acting, main solenoids.

Section 7.3.2.1.2.3.1.5.2.1.10 is replaced in the revised text of the FSAR by 7.3.2.2.2.3.1.10, which reads in part as follows:

> All active components of the primary containment isolation control system, can be tested and calibrated during plant operation with the exception of the main steamline high temperature sensors. By observing the contact action on an HFA type relay during a channel trip condition, the actual drop-out can be verified when de-energized.

Amendment 10, revising Chapter 7, replaced these statements with 7.3.2.1.2.a.10 which in turn references 7.3.2.1.3 of the revised text.

Q. 031.001 (x)

Several of the figures in the FSAR are illegible or are missing component designations and are, therefore, unacceptable. Revise the FSAR to eliminate both illegible and/or unintelligible figures.

RESPONSE:

The FSAR has been revised to eliminate illegible or unintelligible figures.

Q. 031.001 (y)

Clarify the discrepancy in the response time of the reactor core isolation cooling system given in 7.4.1.1.3.1 and that given in 7.4.1.1.3.5 of the FSAR.

Response:

Both 7.4.1.1.3.2 and 7.4.1.1.3.6 have been revised to provide a 30-second response time for the RCIC system.

Amendment 10, revising Chapter 7, replaced both 7.4.1.1.3.2 and 7.4.1.1.3.6 with 7.4.1.1.b.

Q. 031.001 (z)

Clarify the discrepancy between the description in 7.4.1.1.3.5 of the automatic transfer of the suction for the reactor core isolation cooling system which satisfies the design shown in Figure B-25 and the manual transfer shown in Figures 7.4-1a and 7.4-2b which does not satisfy the assumptions made in the Operational Analysis contained in Appendix B.

Response:

The operational analysis in Appendix B (now Appendix 15A) did not assume automatic transfer of the RCIC suction. Section 7.4.1.1.3.5 and Figure B-25 are replaced in the revised text of the FSAR by 7.4.1.1.3.6 and Figure 15.A.6-40 respectively. However, additional discussion about this transfer is provided in the response to Question 031.015.

Amendment 10, revising Chapter 7, replaced 7.4.1.1.3.6 with 7.4.1.1.b.

Q. 031.001 (aa)

Resolve the contradiction between 7.4.1.1.3 and 7.4.2.2.2 so as to provide a clear statement of the conditions under which the isolation valves of the reactor core isolation cooling system will be required to operate and the seismic and environmental conditions for which these valves are qualified.

Response:

Sections 7.4.1.1.3 and 7.4.2.1.2.3.1.4 have been revised to provide a clear statement of the conditions under which the isolation valves for the RCIC will be required to operate. The valves are qualified for operation during those envionmental conditions listed in Table 3.11-2 and Ref 3.11-2. Table 3.2-1 states that seismic classification of the valves to be Category I.

Amendment 10, revising Chapter 7, replaced 7.4.1.1.3 and 7.4.2.1.2.3.1.4 with 7.4.1.1.b and 7.4.2.2.a.4, respectively.

Q. 031.001 (bb,cc)

bb. Clarify the discrepancy between 7.6.1.1.3.1, which states that the refueling interlock system is single failure proof and the design of the reactor manual control system which has a single rod position input path and a single refueling equipment output path.

cc. Modify the FSAR to include concise definitions of the worst case environmental conditions under which the refueling interlocks will be required to operate.

Response:

bb. The refueling interlock system provides two independent channels of instrumentation where either loss of signal or trip signal from either channel will prohibit any further rod movement. The reactor manual control system (RMCS) design has two inputs and the interlock status is merged into a serial data transmission. Even though there is only one rod position path, this information must be in a precise format before it is accepted for any rod movement. If the information is in the proper format, it is then determined if the coded information will allow rod movement. The transmitted information is echoed back by the hydraulic control units and compared to ensure proper transmission. The RMCS provides two refueling equipment output paths. Sections 7.6.1.1.3.1 and 7.6.1.1.3.2 have been revised to clarify the discrepancies.

cc. The refueling platform is not required to operate during the run mode and would not be affected in an accident situation. The worst case environmental conditions under which the refueling interlocks will be required to operate are the normal reactor building conditions listed in Table 3.11-1.

Amendment 10, revising Chapter 7, moved the refueling interlock system discussion from 7.6 to 7.7.1.13.

Q. 031.001 (dd)

Clarify the discrepancy between 7.6.1.6.7.1.1.4 and the design of the reactor manual control system which is presented in 7.7.1.1.

Response:

The isolation, separation, and redundancy features discussed in 7.6.1.6.7.1.1.4 are features of the rod block monitor. The RBM does interface with the reactor manual control system, but is separate from it.

Section 7.6.1.6.7.1.1.4 appears in Rev. 2 of the FSAR as 7.6.1.5.7.1.4. A new subsection has been added which reads:

> "7.7.1.2.3.2.3 Rod Block Interlocks
>
> The rod block functions are discussed in 7.6.1.5.7."

The reference in 7.7.1 to the appropriate subsection for the RMB will eliminate the discrepancy in 7.7.1.

Amendment 10, revising Chapter 7 moved all discussion of the Rod Block Monitor to 7.7.1.8.

BLANK

Q. 031.001 (ee)

Identify and justify all design differences between the reference rod block monitor design described in 7.6-2 and that associated with the new solid state reactor manual control system which is described in 7.7.1.1.

Response:

The reference (Hatch 2) rod block monitor (RBM) described in Reference 7.6-2 and the RBM associated with the reactor manual control system described in 7.7.1.1 are essentially identical.

The comparison of the reference (Hatch 2) and the Zimmer RBM designs were discussed at length between GE and the NRC on April 18, 1977, at GE, San Jose, California. The question is considered to have been resolved for Zimmer.

The designs for the RBM in WNP-2 and Zimmer are identical.

Amendment 10, revising Chapter 7, moved all discussion of the rod block monitor to 7.7.1.8.

Q. 031.001 (ff)

Identify the specific bus which powers the reactor manual control system and the refueling interlock.

Response:

a. The reactor manual control system may be powered from either Instrument bus A or B (PP-7A-A or PP-8A-A). Since the system is not safety-related, the choice of bus has no effect on performance of the system. See 7.7.1.2.2.

b. The busses which supply power to the refueling interlock are 120 VAC instrument busses. See 7.6.1.1.2.

Amendment 10, revising Chapter 7, moved the discussions of the reactor manual control system and the refueling interlocks to 7.7.1.2 and 7.7.1.13 respectively.

Q. 031.001 (gg)

Quantify the recirculation system low water level interlock range, setpoint, and accuracy.

RESPONSE:

The interlock can be set from instrument zero to 60 inches; instrument zero is at 527.5 inches above invert (vessel bottom).

The interlock setpoint is at 31.5 inches above instrument zero.

The accuracy is ± 0.5% of full scale.

Q. 031.001 (hh)

Clarify the discrepancy between 7.7.1.2.3.3.9 and 7.7.1.2.4 of the FSAR with respect to the required motion of the flow control valves under accident conditions.

Response:

The Flow Control Valves are blocked mechanically in the full open position. Therefore, this question does not apply.

Q. 031.001 (ii)

Indicate where in Chapter 7 the information concerning the instrumentation for the reactor building closed cooling water system is located.

Response:

The text of 7.6.1.15 has been added to respond to this question.

Amendment 10, revising Chapter 7, removed discussion of the reactor building closed cooling water system from Chapter 7. The system is described in 9.2.2.

Q. 031.002
(T1.7-1)

Provide process instrumentation and control, logic, wiring, and electrical schematic drawings for: (a) the overpressurization protection (relief) system; (b) the reactor protection system; (c) the leakage detection temperature monitors; (d) the neutron monitor auxiliary trip units; and (e) the reactor protection system instrument racks which are located in the reactor building.

RESPONSE:

The following WNP-2 drawings are applicable:

	System	GE MPL Number	Drawing Identification
A.	ADS Logic (FCD)	B22 1030	731E788
	ADS Elementary	B22 1060	807E180TC
	NBS P & ID	B22 1010	732E103
B.	RPS IED	C72 1010	732E170AD
	RPS Elementary	C72 1050	807E178TC
C.	Leak Detection LED	E31 1010	732E191AD
	Leak Detection Elementary	E31 1050	807E154TC
D.	Neutron Mon. Aux. Trip Schematic	-----	127D1861
	Neutron Mon. Aux. Trip Wiring	-----	195B9206
E.	RPS Inst. Racks (Not in FSAR, but available and auditable at GE)		
	Reac Wtr Lvl and Press Panel Connection	H22 P004	127D1827TC
		H22 P005	127D1827TC
	Reac Wtr Lvl and Press Panel Connection	H22 P026	828E390TC
	Reac Wtr Lvl Press Panel Connection	H22 P027	828E387TC

Q. 031.003
(3.1.2.2)

Describe the provisions for meeting the inspection requirements of General Design Criterion 18, as requested in 3.1 of the Standard Format.

RESPONSE:

Inspection and testing of the Electric Power System, including testing of the standby diesel generators is discussed in 3.1.2.2.9.1.

Q. 031.004 (RSP)
(3.10.1.2)

It is the staff's position that Class 1E equipment must be seismically qualified for proper operation prior, during and after a seismic event. (Refer to Section 3 of IEEE Std. 344-1971). Since proper operation does not include spurious operation, we require you to: (a) identify each piece of Class 1E equipment which was not tested for spurious operation during seismic testing; (b) identify each non-Class 1E device (such as fire protection systems) which could adversely affect Class 1E equipment if it were spuriously actuated and the Class 1E equipment so jeopardized; and (c) provide the results of an analysis of all possible combinations of spurious operations listed in items (a) and (b) and their effect on the safe shutdown capability, assuming a loss of offsite power.

Response:

a. All pieces of Class 1E equipment which were seismic tested were also tested for spurious operation. Qualification acceptance required freedom from spurious operation during testing.

b. 1) There is no GE supplied non-Class 1E equipment which, if it were spuriously actuated, could adversely affect Class 1E equipment.

   2) Balance of plant non-Class 1E equipment which could adversely affect Class 1E equipment is discussed in 3.10.1.2 and Table 3.10-5.

c. No combinations of spurious operations were identified in (a) and (b) and, hence, there is no effect on the safe shutdown capability, assuming a loss of offsite power.

See also revised 3.10.1.2 and Table 3.10-5.

Q. 031.005 (RSP)
(3.10)
(3.11.3)
(7.2.2.2)

A request for documentation of the seismic and environmental qualification of Class 1E equipment is contained in 3.10 and 3.11 of the Standard Format. This request is applicable to all engineered safety features, reactor protection systems and all supporting systems. It is not limited to those particular safety-related supporting systems supplied by General Electric. Accordingly, we required you to provide the information requested in 3.10 and 3.11 of the standard Format for all Class 1E systems in accordance with: (a) the NRC staff positions stated in Attachments 1 and 2: (b) IEEE Std. 323-1971; and (c) IEEE Std. 344-1971. Identify and justify any exceptions.

Response:

The requested seismic and environmental qualification data for Class 1E equipment for all engineered safety features, reactor protection systems and all support systems is provided in the revised text and tables of 3.10 and 3.11, and in the Environmental Qualification Report and Dynamic Qualification Report referenced therein.

For additional discussion refer to 7.2.1.2.7 (RPS), 7.3.1.1.1 (ECCS), 7.3.1.1.2.12 (PC and RVIS), and 7.3.1.4.11 (RHR/ Containment Spray).

Additional discussion of conformance to IEEE 323-1971 is provided in 7.2.2.1.2.3.4 (RPS), 7.3.2.1.2.3.5 (ECCS), and 7.3.2.2.2.3.5 (PC and RVIS).

Amendment 10, revising Chapter 7, references Tables 3.11-1 and 3.11-2, Ref 3.11-2 and Section 6 of Appendix J for environmental qualification for all engineered safety features, reactor protection systems and all support systems. Section 7.1.2.3 discusses conformance to IEEE Standards 323-1971 and 344-1971 which were reviewed in late 1982 and early 1983 by NRC SQRT and Environmental Audits. The programs are described in the reports referenced above. Equipment qualification adequate for power operation was established prior to fuel load in December 1983. An ongoing program exists to complete qualification of equipment justified for interim operation and to respond to proposed plant changes and, therefore, to maintain plant equipment qualification documentation. Amendment 10 additionally replaced 7.2.1.2.7, 7.3.1.1.2.12, and 7.3.1.1.4.11 with 7.2.1.1, 7.3.1.1.2 and 7.3.1.1.4, respectively.

Q. 031.006
(3.11)

The staff requests that the following information regarding the qualification test program be provided for Class 1E equipment: (a) the equipment design specification requirements; (b) the test plan; (c) the test setup; (d) the test procedures; (e) the acceptability goals and requirements; and (f) the test results.

Provide this information for each of the following Class 1E components: (a) the 4.16 kV switchgear SM 7; (b) the damper operator for WMA-V-52C; (c) the fan WMA-FN-52B; (d) the logic equipment for the standby gas treatment system (e) the diesel-generator control equipment; (f) the 480 V ESS switchgear MC-7A-A; and (g) the solenoid valve for the main steam line isolation valves.

Response:

The seismic and enviornmental qualification program was fully described in the "WNP-2 Environmental Equipment Qualification Report for Safety-Related Equipment", referenced in 3.11 and the "WNP-2 Dynamic Qualification Report for Safety-Related Equipment", dated September 1982.

This document formed the basis for the NRC Seismic Qualification Review Team and Environmental Audits of the Supply System program. During these audits the information provided included the design specification requirements, test plan, test setup, procedures, acceptability criteria, and test results. These were provided for Class 1E components selected by the NRC auditors. Similar equipment to that requested by this question was audited by the NRC. Therefore, a separate submittal of the backup documentation requested by this question will not be made. Summary information on the requested equipment may be found in the previous mentioned reports.

The documentation will be reviewed to insure that the testing was adequate to meet the seismic and environmental extremes under which the equipment must either function or not fail.

A composite list was included in the FSAR as equipment tables in 3.10 (seismic) and 3.11 (environmental). This data is now captured in an on-line computer data base called the Master Equipment List (MEL).

The extensive review program underway will also satisfy the requirements of IE Circular 78-08, address the degree of compliance with NUREG-0588, and establish the conservativeness of seismic tests and analysis performed to IEEE-344, 1971.

The detailed results of this review will be made available to NRC SQRT and environmental review personnel during their site documentation reviews.

The Supply System feels this consolidated qualification review of all safety-related equipment encompasses the scope of the concerns of this question and further NRC queries should be held in abeyance until the entire program can be reviewed in the context of the new 3.10, 3.11 and the NRC audits.

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Q. 031.007
(1.2.2.5)
(3.1.2.3)
(4.6)

Chapter 4 (and more specifically, 4.6) of the FSAR is incomplete. Provide a description of the standby liquid control system which is referred to in 1.2.2.5.20, 3.1.2.3.7.1, and 7.4.1.2.

RESPONSE:

The description of the standby liquid control system is provided in 9.3.5. 4.6 references 9.3.5.

Q. 031.008
(4.4.3.3)
(F5.4-2a)

The design of the recirculation system as described in 4.4.3.3.3 and 5.4.1.3 of the FSAR appears to be obsolete. It is the staff's understanding that this General Electric design does not include a bypass in the recirculation system piping. Accordingly, we request that you:

1. Clarify the discrepancies between the cited sections of the FSAR and Figure 5.4-2a which doesn't show a bypass; and

2. Revise the FSAR, where appropriate, to describe the actual design of the WNP-2 recirculation system.

RESPONSE:

4.4.3.3, 5.4.1.3 and 5.4.1.4 have been revised or added to incorporate the response to this question.

Q. 031.009 (RSP)
(5.2.2.1)
(7.3)
(15)

The overpressure relief operation is listed as a protective function of the safety/relief valves in 5.2.2.1.1 and 5.2.2.1.3 of the FSAR. Further, the Chapter 15 analyses are predicated on the proper functioning of these valves. On this basis, the staff concludes that these valves are an engineered safety feature and must meet the requirements for a Class 1E system. However, the present design does not satisfy the single failure criterion and is, therefore, unacceptable. Accordingly, we require you to provide a revised design which will satisfy the single failure criterion. Revise the FSAR in accordance with the request for information continued in 7.3 of the Standard Format, including the following topics: (a) redundancy in electrical power sources; (b) pressure switch setpoints, accuracy and range; (c) redundancy in valve operating energy sources; (d) seismic qualification; and (e) quantification of the minimum number of valves which must function to limit to a safe value, the pressure spike resulting from a spurious main steam line isolation valve trip at full power at the worst time in core life.

RESPONSE:

The overpressure relief operation discussed in 5.2.2.1.1 and more specifically in 5.2.2.1.3 utilizes the self actuated safety function of the valve.

In order to describe the present plant design and to respond to requested information relative to the design, the following introductory discussion is presented prior to the direct reponse.

Introduction

The subject equipment of this inquiry involves the so-called "safety/relief valves". These multi-use, multi-function components perform three distinct, separate plant operational functions as follows:

1. The "safety" function provides Reactor Coolant Pressure Boundary (RCPB) overpressurization protection by rapidly reducing the energy content of the system.

2. The "ADS" function provides an alternate means of rapidly depressurizing the RCPB to allow low pressure injection of cooling water into the reactor vessel.

3. The "relief" function provides anticipated relief action to soften transient effects by maintaining steady state operating conditions inside the RCPB.

Functions 1 and 2 above are considered safety functions; functions that provide protection of the health and safety of the general public by assuring that the reactor core is maintained in a cooled and floodable volume, that the containment integrity is maintained, and that the reactor can be safely shut down. These apply to the "Design Basis Accident" and the "Special Event" described in Chapter 15 and Appendix 15A.

Function 3 above is considered a power generation function; a function that provides equipment operating conditions conducive to continued, reliable performance. This is accomplished by reducing or minimizing the above-normal thermohydraulic or thermodynamic effects of anticipated operational transients. This applies to "Normal Operation" and "Anticipated Operational Transients" cited in Chapter 15 and Appendix A.

Failure Analysis Considerations

Failure analysis of the Design Basis Accidents (DBA) and Special Events, in addition to the "free" initial events (e.g., LOCA), include the application of Single Active Component Failure (SACF), credit for only safety grade equipment for safety functions, natural phenomena effects, and automatic safety actions.

But failure analyses of the "Normal Operations" and "Anticipated Operational Transients" involve only the single component failure (SCF) or the single operator error (SOE) as the initiating event. There are no further requirements such as additional SACF, or use of only safety equipment, or only automatic action.

Considerations of superposition of the SACF, etc., in addition to the SOE or SCF related initiating event, fall into the frequency of occurrence classification for "Abnormal Operational Transients", "Design Basis Accidents", or "Special Events" categories described in Chapter 15 and Appendix 15A. These classifications allow the previous normal operation objective, that there should be no loss of fuel cladding integrity, to be relaxed to permit fuel performance at MCPR < 1.0, and of course dose impacts greater than 10CFR20.

Design Basis Aspects

1. The "safety" function is performed by safety grade, seismically and environmentally qualified parties of the safety/relief valves. These devices are self-actuated.

2. The "ADS" function is performed by safety grade seismically and environmentally qualified portions of selected safety/relief valves. The circuitry and controls for these devices have equal qualifications. Although the ADS safety function is redundant to the HPCS in the ECCS network, some additional redundancy was added, such as redundant power supplies, and an additional valve.

3. The "relief" function is performed by similar circuitry and control to the ADS function, but since it is not a safety function, an official quality control certification of qualification is not provided. The valves are powered from a single battery source. Each valve is, in itself, independent from other valves. The only common mode connections to all relief valves are the control and the power source.

The battery source, although not in this connection redundant, is safety grade. It is supplemented by a battery charger, and it supplies only selected loads which are continuously monitored and tested by ESF-type technical specification requirements. All of the battery source equipment is housed in seismically and environmentally qualified structures.

Any failure mode and effects analysis of the relief valve function aspects which could negate the relief function, would involve at least an abnormal incidents classification or possibly an accident event classification. Under these conditions, the relief function would not be a safety design basis required function.

As stated above, the only postulated single failure in addition to the initiating event (example: turbine trip), which would negate the "normal operation" of the relief valves is the single source battery failure. It being an ESF component, its failure is extremely remote. In any case, the protection of the RCPB from overpressurization would be precluded by the safety valve function of the safety/relief valves.

In summary, the present design basis and interpretation cited above is consistent with previously reported and justified safety/relief valve classifications and interpretations of the GDC related to accident and transient occurrence protective measures.

See also revised 5.2.2.1.4.

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Q. 031.010
(5.4.1.3)
(7.7.1.2)

Provide the following information regarding the recirculation flow control system:

1. Describe the safety-related consequences of cavitation in any of the recirculation pumps, jet pumps, and flow control valves.

2. Provide justification for the design lifetime of 40 years, with only replacement of the valve packing, for the components cited in Item (a) in light of the consequences of cavitation. Provide justification for the use of non-Class 1E interlocks to preclude cavitation.

3. Describe the consequences of a failure of either the valve position interlock or the thermal power interlock to prevent a transfer of the pumps from low to high speed.

4. Describe the consequences of a failure of the motion inhibit interlock during a postulated loss-of-coolant accident.

5. Explain why it is necessary to provide a high gain output from the flux controller when the core is sensitive to flow changes in the range of 0.015 to 0.15 Hz.

6. Explain why it is necessary to provide a low signal level limit when the flux controller is provided with amplification.

7. Describe the relay circuits which bypass the limiter and quantify the setpoints and conditions which cause the limiter to be bypassed; discuss the term "desired position" and explain why the limit function is needed.

RESPONSE:

(a) Recirculation System Cavitation

Recirculation Pump

Low NPSH would produce vapor bubbles near the center of the impeller. These would collapse as they move toward the impeller exit. Consequently, cavitation affects the impeller surfaces and not the bowl or discharge region.

Flow Control Valve

Low NPSH would produce vapor bubble in the fluid stream before it reaches the valve ball due to the decrease in static pressure. These bubbles would collapse in the pressure recovery areas at the ball and valve exits affecting material in these areas.

Jet Pump Nozzle

Vapor bubbles could form in the nozzle as the pressure decreases and velocity increases toward the nozzle exit. These would collapse in the pressure recovery area in the jet throat, affecting material in the throat area.

Jet Pump Suction

Vapor bubbles could form in the suction flow as pressure decreases and velocity increases from the downcomer annulus to the throat. These would also collapse in the throat pressure recovery area.

Cavitation Coefficients

The recirculation pump, jet pump, and flow control valve are tested to determine their cavitation coefficients so that prolonged operation in cavitating regimes can be avoided.

NPSH

System NPSH depends on reactor water level and water subcooling below vessel dome saturation temperature. During normal operation, feedwater flow provides the most subcooling by mixing with the steam separator return flow making the downcomer water temperature less than saturation temperature.

Equipment Damage Provisions

Cavitation interlocks for the recirculation pump, flow control valve and jet pumps; since cavitation produces material damage after long-term operation, and the damage potential decreases with an increase in water temperature, short periods of cavitation during a transient or accident are not a concern. However, long-term operation that might occur due to delayed operator action to trip the pumps is of sufficient concern to call for inspections during the next refueling outage. Consequently, to avoid the need for such inspections, automatic interlocks are installed.

(b) The consequences of cavitation would require inspection of the affected component and repair or replacement if the inspection showed unacceptable damage. Consequently, cavitation could call for increased scheduled outage time for inspection/repair affecting plant availability power generation design goals. Class 1E equipment is not necessary for power generation design requirements.

(c) This event requires two failures; an operator error and an interlock failure. Consequently, it is considered an accident and clearly is bounded by the LOCA.

(d) The motion inhibit interlock was deleted from the design since it is not needed. (Also see response to Question 031.001 (e)).

(e) The flux controller supplies a total drive flow demand signal to a flow controller station, which in turn supplies each flow loop with a demand signal. Under automatic control, the flux controller output is compared to the sensed loop flow from the feedback proportional amplifiers in each loop. The error signal is fed via the flow controller amplifier with the feed forward drive flow demand signal to the valve position, resulting in a change of loop flow and therefore core power. (See 7.7.1.3.3.4.3).

Neutron flux is sensitive to changes in core flow in the frequency range of approximately 0.015 to 0.31 Hz. The flux controller is a lag/lead compensated proportional-integral (P-I) controller. The controller provides a high-gain output for low-frequency input signal from feedwater or pressure disturbance.

(f) The drive flow demand limiters are adjustable. The high signal limiter is to establish the maximum drive flow demand limit needed for the upper end of the automatic load-following range. The low signal limit is determined from a core stability criterion and defines the lower end of the automatic load-following range. There is no flow limit, and the valve can be closed to its minimum position when the master controller is in manual mode operation. (See 7.7.1.3.3.4.4).

(g) A limiting function is provided for one feedwater pump trip flow runback. An electronic limiter with reasonable range adjustment is provided in each main flow control loop. This limiter is normally held bypassed by auxiliary devices such as relay contacts. When one feedwater pump trip coincides with reactor low water level alarm, the main regulating valve control signal is limited to close the valve to the desired position. (See 7.7.1.3.3.4.8).

See also 5.4.1.3.1.

Amendment 10, revising Chapter 7, deleted 7.7.1.3.3.4.3, 7.7.1.3.3.4.4 and 7.7.1.3.3.4.8. The revised text references Appendix H for the discussion of the recirculation flow control system.

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Q. 031.011
(6.2.2.3)

The description of operator actions is incomplete. Provide the following information: (a) describe the operator actions which are necessary to establish containment spray; and (b) describe the design provisions, if any, which prevent the operator from shutting down a pump too early or diverting flow too soon.

Response:

(a), (b) Discrepancies have been present in the WNP-2 FSAR (see Question 031.001 (q)) in describing the operation of the containment spray system. These discrepancies have been corrected. One can now obtain the operator actions, procedural controls and design provisions for containment spray operation in 6.2.2.2, 6.5.2.2 and 7.3.1.1.4. These sections have been revised. Figure 7.3-16 has been updated to show the correct initiation signals for containment spray.

Amendment 10, revising Chapter 7, replaced 7.3.1.1.4 with 7.3.1.1.4.b.

Q. 031.012
(6.2.4.2)

Describe the pump motive source which is used to provide feedwater flow after the main steamline isolation valves close and which forms the basis for the assumptions in 6.2.4.2.1.2 of the FSAR.

RESPONSE:

The text of 6.2.4.3.2.1.1.1 has been revised to incorporate the response to this question.

Q. 031.013 (RSP)
(6.3.1.3)
(7.3.1.1)

Provide justification for not testing the emergency core cooling system flow rate and the associated sensing networks during normal operation. Define the term "sensing network" as used in 6.3.1.3 of the FSAR. Identify each network which cannot be tested during normal operations. It is the staff's position that the WNP-2 design should provide engineered safety feature circuits which satisfy the guidance contained in Regulatory Guide 1.22. Accordingly, we require you to provide a revised design which conforms with the staff's position on this matter and to provide a description of these sensors and networks which provides the information requested in 7.3 of the Standard Format. Clarify the discrepancy between 6.3.1.3 and 7.3.1.1.1.2.1.2 with regard to the testability of the emergency core cooling system.

Response:

Section 6.3.1.3 was in error. The text of 6.3.1.1.2(m) and 7.3.2.1.2.3.1.10 have been revised to reflect that all active components of the ECCS are testable during normal operation.

Amendment 10, revising Chapter 7, replaced 7.3.2.1.2.3.1.10 with 7.3.2.1.2.a.10 which references 7.3.2.1.3.a for the discussion of ECCS testability.

Q. 031.014
(6.3.2.2)
(6.3.2.8)
(F6.3-1a)

The location of sensors LS N001 A, B, C, and D, as shown in Figure 6.3-1a, does not appear to meet Seismic Category I requirements. Revise the design of WNP-2 to assure that the sensors controlling the transfer of suction to the suppression pool will be seimsically and environmentally qualified for their location and environment.

Response:

Condensate storage tank level monitoring switches are mounted on a Class 1 standpipe in the reactor building and designed and qualified to Seismic Category I requirements and are environmentally qualified. For further details see responses to Questions 031.128, 211.146, and 211.197.

Q. 031.015 (RSP)
(6.3)
(7.4.1.1)
(7.4.2.2.2)
(FB-25)

In accordance with the assumptions and results of the operational analyses contained in Appendix B, it is the staff's position that the reactor core isolation cooling system (RCIC) is an engineered safety feature. Accordingly, we require you to:

(a) Provide a revised design which will satisfy the requirements for an engineered safety feature and document the revised design in the manner requested in 7.3 of the Standard Format.

(b) Provide an initiation logic which includes manual initiation of the RCIC and system isolation valves at the systems level in accordance with 4.17 of IEEE Std. 279-1971.

(c) Demonstrate how the revised design satisfies the following design criteria:

(1) IEEE Std. 279-1971; (2) IEEE Std. 323-1971; (3) IEEE Std. 338-1971; (4) IEEE Std. 344-1971; and (5) the staff position stated in Attachment 2 to this enclosure.

(d) As an alternative response to items (a) through (c), provide an analysis which demonstrates that the automatic depressurization system, in conjunction with the low pressure coolant injection system and/or the low pressure core spray system, can provide emergency core cooling in the event of a control rod drop accident. This analysis should assume a failure of the high pressure core spray system to function and incorporate a 120 second delay in initiation of the automatic depressurization system.

RESPONSE:

(a) The present design of the RCIC System is functionally the same as that approach at the time of the issuance of the construction permit for WNP-2. In addition, it is the same design as the one licensed on other plants such as Hatch 1, Brunswick 1 and 2, et al, which have recently received operating licenses. The RCIC is designed primarily to maintain sufficient cooling in the reactor vessel in case of an isolation with a loss of main feedwater flow. It is presently classified as a safe shutdown system, and should not be classified with the HPCS System in the ECCS.

The design requirements for the system and associated instrumentation are described in 7.4.1 and 7.4.2. These requirements are such that the RCIC can be used in conjunction with the HPCS and together with the latter meets single failure criteria in mitigating the consequences of the design basis control rod drop accident. This is the only accident for which credit for automatic action of the RCIC is taken. This hypothesized accident has no associated seismic event. However, the equipment and component meet seismic classification (Category 1) as defined in Table 3.2-1. The turbine has been fabricated to the applicable Code requirements described in Note 13 of Table 3.2-1.

Additionally, the RCIC System may provide the ability to mitigate the consequence of small pipe breaks, but it has not been provided for such purposes. The ECCS provide redundant protection for the entire spectrum of pipe breaks as shown in 6.3.

Based on the above discussion and review and construction schedules for WNP-2, no design change or reclassification is necessary.

However, in order to automate the manual transfer aspects of RCICs water source in the unlikely event the condensate storage tank inventory is unavailable for use if called upon, the applicant is modifying the present design to include the following features:

1) Automatic transfer circuitry equivalent to HPCS auto transfer system will be provided.

2) Condensate storage tank site natural phenomena considerations will be taken into account in order to assure that the automatic transfer function is not negated.

The above plant modification is similar to that proposed for the Zimmer plant.

(b) The RCIC System has automatic initiation and isolation, and manual initiation and isolation. Compliance with IEEE 279-1971, paragraph 4.17, is in 7.4.2.1.2.3.1.17.

(c) The design compliance for the present system is presented in 7.4.2.1.

(d) Since a, b & c have been responded to, a response here is not applicable.

Amendment 10, revising Chapter 7, describes the RCIC system in 7.4.1.1; 7.4.2.1.2.3.1.17 was replaced by 7.4.2.2.a.17. Section 7.4.2 of the revised text replaced 7.4.2.1.

Q. 031.016 (RSP)
(6.3.3.9)
(7.0)
(7.2)
(15)

The discussions of response times and the testing of response times in 6.3.3.9, 7.2 and elsewhere in Chapter 7, are inadequate. Chapter 15 does not provide the response times, accuracies nor ranges for instrumentation and control systems which were assumed in the accident analyses. It is the staff's position that the design of systems which are required for safety shall include provisions for periodic verification of the minimum performance of instruments and controls that are not less than those assumed in the safety analyses. The bases for this position are General Design Criterion 21, 3.9 of IEEE Std. 279-1971, and IEEE Std. 339-1971. Accordingly, we require you to demonstrate:

a. The capability to periodically verify the minimum performance characteristics of all systems, including the appropriate procedures, which are required for safety. Testing shall include the entire system from, and including, sensor to actuator output.

b. Compliance with Branch Technical Position 24 of Appendix 7-A of the Standard Review Plan.

c. Compliance with the requests for information in 15 of the Standard Format.

RESPONSE:

a. The Technical Specifications for WNP-2 will require periodic response time testing of the C & I protection and safeguards systems from sensor through actuation systems. The procedures for initial system test and periodic surveillance testing will be developed (see 14.2.12.1.18 and Chapter 16) by WPPSS.

b. The requirements noted in a. comply with Branch Technical Position 24 of Appendix 7-A of the Standard Review Plan.

c. Chapter 15 is not the appropriate location for sensor response times, accuracies, and ranges for instrumentation and control systems. Tables 7.2-1, 7.3-1, 7.3-2, 7.3-3, 7.3-4 and 7.4-1 have been revised to provide instrument ranges on reactor protection systems and ESF systems instrumentation. Trip settings and response times used in Chapter 15 analysis are contained in the WNP-2 Technical Specifications".

Amendment 10, revising Chapter 7, changed Tables 7.3-2, 7.3-3 and 7.3-4 to 7.3-3, 7.3-5 and 7.3-7 respectively.

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Q. 031.017 (RSP)
(F 6.3-10)
(F 6.3-11a)
(F 6.3-11b)
(R 6.3-7)

We have concluded that NEDO-10739 is unacceptable as a basis for licensing. Our position on this matter is given in Attachment 3. With respect to the WNP-2 reliability studies, we require you to:

a. Provide the details of the program to select and analyze the available failure rate data, reliability analyses, and operating experience of equipment identical to that which will be used in the WNP-2 facility when developing your functional testing and calibration schedules.

b. Describe how the analyses of the availability of the reactor core isolation cooling system, the main steamline isolation valve leakage control system, the rod sequence control system, the reactor manual control system, and the turbine governor system were performed.

c. Describe the function of Figures 6.3-10 and 6.3-11, including sufficient information to permit the staff to interpret these figures. Identify the success paths, goals, failure rates and repair times.

RESPONSE:

a. The functional testing and calibration of the ECCS will be performed in accordance with the schedules established in Appendix A, Technical Specifications. These schedules will be based on the requirements defined in the Standard Technical Specification for BWR's.

b. There are no quantitative reliability analysis for the RCIC, the MSIV-LCS, the RSCS, the reactor manual control system, or the turbine governor system. If quantitative analysis were to be performed, the guidelines of IEEE 352-1972, "General Principles of Reliability Analysis of Nuclear Power Generating Station Protection Systems" would be applied.

c. With the incorporation of 10CFR50.46 wherein successful operation of the ECCS depends on multiple systems and a single failure type of analysis is performed, there is no longer validity in Figures 6.3-10 and 6.3-11. These two figures were generated at the time of the ECCS Interim Acceptance Criteria. Single failure analysis are now performed per 10CFR50.46 to demonstrate the adequacy of the ECCS.

See also 6.3.4.2 and 6.7.3.1.

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Q. 031.018 (RSP)
(6.3.3.9)
(7.2.1.1)
(7.3.1.1)
(F6.3-1)

It is the staff's position that the use of safety-related equipment which is not seismically and environmentally qualified, is a violation of the requirements of General Design Criterion 2. Accordingly, we require you to provide modified designs for all safety systems which do not require sensors or equipment located in non-seismic Category I structures. Alternatively, identify a reactor power limit below which all such equipment and sensors are not required.

RESPONSE:

This same concern was expressed by the NRC on the GESSAR docket STN50-447. GE responsed with an analysis of 1) the probability of an unsuccessful trip path, and 2) reactor system thermal hydraulics capability. The probability analysis was submitted as Attachment A to the response to Q. 2.2.2.22 on the GESSAR docket. The results are described in paragraph 3 on page R222.22-2 of the docket.

In the response, the General Electric Company stated that the probability of losing these inputs to the reactor protection system was extremely small, on the order of 10^{-6} per year. The General Electric Company also investigated the percentage of fuel rods in the core which are subject to boiling transition prior to termination of the transient by the backup scram (high flux). The most limiting case was that for turbine trip with a backup scram in which the turbine bypass system failed to operate. For this case, the General Electric Company indicated that 7% of the fuel rods in the BWR/6 core would experience boiling transition.

The NRC evaluated the General Electric Company GESSAR response and concluded that it was adequately established that this is an extremely low probability event and that effects of the transient were conservatively calculated. The NRC used the assumptions of the control rod drop accident to provide a very conservative estimate of the offsite dose consequences. It was assumed that those fuel rods which experienced boiling transition perforate. The quantity and behavior of fission products which are released were treated in the same manner as were the control rod drop accident.

Based on the above assumptions, it was concluded by the NRC in the GESSAR Safety Evaluation Report that the probability of the event was low enough to permit use of the conservative accident assumptions and compare the offsite doses to 10CFR100. It was concluded that the offsite doses are well within 10CFR100 limits.

An analysis for Zimmer 1, the lead BWR/5, has concluded that only 5.0% of the fuel rods would experience boiling transition in the limiting case.

A WNP-2 analysis would be similar to the GESSAR and Zimmer results, and lead to the same conclusion. It is, therefore, not planned to modify the design nor perform a WNP-2 analysis, nor is it believed that a power limit is required.

Q. 031.019
(7.1.2.1)

Demonstrate how the design of the main steamline isolation valve leakage control system satisfies the requirements of 4.19, 4.20, and 4.21 of IEEE Std. 279-1971.

Response:

The main steamline isolation valve leakage control system satisfies the requirements of IEEE Std. 279-1971, 4.19, 4.20, and 4.21 as follows:

Identification of Protective Actions (IEEE 279-1971, paragraph 4.19)

Initiation of the MSLIV-LCS is indicated in the control room.

Information Read-Out (IEEE 279-1971, paragraph 4.20)

Meters located in the control room provide indication of process variables necessary for the proper operation of the MSLIV-LCS. Indicator lights actuated by valve position switches provide valve position indication.

System Repair (IEEE 279-1971, paragraph 4.21)

The system is designed to provide easy recognition of malfunctioning equipment through proper test procedures. Accessibility is provided for the sensors and controls to facilitate repair or adjustment. MSLIV-LCS isolation valves are located in the steam tunnel and repair is made during shutdown.

See also 7.3.2.3.2.3.1.19, 7.3.2.3.2.3.1.20 and 7.3.2.3.2.3.1.21.

Amendment 10, revising Chapter 7, replaced 7.3.2.3.2.3.1.19, 7.3.2.3.2.3.1.20 and 7.3.2.3.2.3.1.21 with 7.3.2.1.2.a.19, 7.3.2.1.2.a.20 and 7.3.2.1.2.a.21 respectively.

Q. 031.020
(T7.1-1)

Table 7.1-1 is incomplete. Complete this table by filling in all of the blank spaces so as to indicate if the cited systems are a design which is unique to the WNP-2 facility or is similar to those used on other nuclear power plants.

RESPONSE:

Table 7.1-1 has been replaced by revised Table 7.1-2.

Q. 031.021
(7.2.1.2)
(T7.2-1)(T7.3-1)
(T7.3-3)(T7.3-5)
(T7.3-7)(T7.4-1)

The information which is presented in Table 7.1-6 and 7.2.1.2.9 is inadequate. The trip settings and margins for some instruments are missing as well as the range and accuracy for other instruments. Additionally, the units of measurement for other instruments are missing while the units of measurements for other instruments are not consistent with the units for the setpoint. Provide the following information:

a. All trip settings which are required for safety, expressed in units which are consistent with the instrumentation range.

b. All ranges and accuracies for all instrumentation systems which have a safety function or which provide required safety system support.

c. The design criteria used in establishing the required range of instruments in the reactor protection systems, engineered safety feature systems, and other safety-related systems.

d. The design criteria used in determining which portion of the range of an instrument may be used for automatic initiation of a protective function.

e. Where trip settings are to be based on operating experience, state the initial values which are to be used and provide the bases.

f. Response times, provide both required and calculated.

Our concern is that, in previous designs of nuclear power plants, the setpoints have either drifted beyond the range of operability of the sensor or sensor foldover has occurred because the setpoints were initially set too close to the extreme ends of the instrument range.

Response:

The Tables provided for the following answers include instruments in addition to those required for safety.

(a, b, and f) The requested information is contained in Tables 7.2-1, 7.3-1, 7.3-3, 7.3-5, 7.3-7, 7.3-9 and 7.4-1.

Tables 7.2-1, 7.3-1, 7.3-3, 7.3-5, 7.3-7, 7.3-9, and 7.4-1 have been revised to provide nominal trip settings, instrument ranges, accuracy, and response times. Margin will be provided later, as available.

(c) The range for safety-related instrumentation is selected so as to exceed the expected range of the process variable being monitored. This may, as in the case of the neutron monitoring system, require more than one instrument to cover the expected range.

(d) The trip setpoint is located in that portion of an instrument's range which provides the required accuracy. For example, although the drywell high pressure setpoint is near the low end of the instrument's range, setpoint drift downward would be in a conservative direction. Also, this instrument has been proven in other BWR's.

(e) Initial trip setting values are established from operating experience with similar size plants, and backed up with analysis as necessary. The initial values are the trip settings listed in the tables referenced in (a, b, and f).

See the response to Q. 031.037 for a discussion of setpoint drift. Revised 7.1.2.4 and the response to Q. 031.001(r) provide additional discussion regarding this question.

Amendment 10, revising Chapter 7, replaced 7.1.2.4 with 7.1.2.5. Table 7.1-6 has been deleted.

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Q. 031.022
(7.2.1.1)

Section 7.2.1.1.3.1.1 of the FSAR is incomplete. Provide a description of the source range trips which provide rod block and scram protection for the initial startup of new cores.

Response:

In addition to 7.2.1.1.4.2 the source range trips which provide rod block and scram protection for the initial startup of new cores are described in 7.6.1.6. Functions for rod blocks are provided in 7.6.1.6.2. Figures 7.6-15a, 7.6-14 and 7.2-1 also provide descriptive information.

For the initial fuel load, high-high trip contacts from each SRM are combined to produce a 1:4 non-coincident reactor trip through the manual scram portion of the circuit. Following the initial fuel loading and startup, these latter contacts are permanently shorted to remove the SRM reactor trip function.

Amendment 10, revising Chapter 7, replaced 7.2.1.1.4.2 with 7.2.1.1.b.1, the rod block monitor protection is discussed in the revised text 7.7.1.7. Table 7.7-4 provides SRM trips.

Q. 031.023
(7.2.1.1)

Provide the design criteria and a description of the scram discharge volume switches and their qualifications testing in accordance with the request for information contained in 7.2 of the Standard Format, including the following information: (a) the manufacturer, (b) the type of float (self equalizing or sealed); (c) the float material and magnet material; and (d) the following qualification test conditions, including (1) water temperature; (2) pressure; (3) duration of test conditions; (4) number of test cycles; (5) period between test cycles; (6) extremes of external temperature, pressure, and humidity; and (7) radiation source, strength, and dose.

RESPONSE:

The mechanism of this level switch is a float which utilizes a non-self-equalizing float. There are four switches mounted on the scram discharge volume. The design pressure and temperature are 1250 psig and 500°F respectively. Additional information on design specifications and qualification test conditions are as follows:

(a) Manufacturer: Magnetrol

(b) Type of Float: Non-self-equalizing float

(c) Float Material: 347 SS
Magnet Material: ALNICO 5

(d) No test data available; is a common item list:

150°F maximum temperature
100% RH
1.1×10^{7}
$\pm$ 1/4 in accuracy

Q. 031.024
(7.2.1.1)
(7.2.2.2)

The discussion of the turbine stop valve closure and turbine control valve fast closure scrams in 7.2.1.1.5.4.4, 7.2.1.1.5.4.5, and 7.2.2.2.2.2.1.1.2.1 indicate that these scrams are "...required to provide a satisfactory margin to core thermal-hydraulic limits." It is also indicated that these scrams provide additional margin over the high-pressure scram to preclude overpressurizing the nuclear system although the nuclear steam high-pressure scram, in conjunction with the pressure relief system, is adequate. In view of the present location of the sensors for the turbine stop valve closure and the turbine control valve fast closure scrams in a non-seismic Category I structure, we require you to provide the results of an analysis of the effects of a total loss of these two scram signals during a turbine/generator trip or an analysis of the adequacy of the other protection system signals to provide a satisfactory margin for thermal-hydraulic limits in the core.

RESPONSE:

Operability of the anticipatory signals from the turbine governor valve fast closure or turbine stop valve closure following a safe shutdown earthquake is not a system design basis. There is no reason to expect concurrent failures of these trips without the safe shutdown earthquake. However, if the gross failure of these trips should occur, the reactor would scram on high neutron flux. The results of this event would not be more severe than the one caused by closure of all the main steamline isolation valves without MSIV position switch trip. That event is discussed in 5.2.2.2.2.4 as the relief valve sizing transient.

Q. 031.025
(7.2.2.2)
(7.5.1.4)

The discussion of shutdown, isolation and core cooling indication is inadequate. Provide the following additional information:

a. Identify which instrument bus supplies power to the control rod status lamps.

b. Identify which instrument bus supplies power to the control rod scram pilot valve status lamps.

c. Justify the use of the power range channels and the downscale indication of the recorders as a valid indication of reactor subcriticality following a loss of offsite power.

d. Identify the annunciators which have a safety-related function.

e. Describe the qualifications of the annunciators, and the plant process computer. Demonstrate that they satisfy the requirements of General Design Criteria 13, 20(2), 21, 22, 23, and 24 and IEEE Stds. 279-1971, 323-1971, 338-1971, and 344-1971. If these criteria are not met, provide justification for the use of this equipment in safety-related systems.

RESPONSE:

a. Power for the control rod status lamps is fed from instrument bus 1A.

b. Electric power for the control rod scram pilot valve status lamps is fed from instrument bus 1A.

c. There is no requirement for the power range channels and recorders downscale indication to remain available to the operator following a loss of offsite power. A loss of offsite power would result in all scram valve solenoids de-energized and scram.

d.,e. The function of the annunciators is to supply information to the operator. They are not protective systems in that they do not provide any trip signals.

Annunciators are only one means by which the operator can assess the status of a system. It should be recognized that no operator action is required for the first 10 minutes following an incident. This gives the operator adequate time to review system status from operating lights, indicators, and relay positions.

The annunciators and plant process computer provide information to the operator. Neither provides any trip signals. The process computer provides thermal hydraulic information to the operator which he uses to keep the plant operating within technical specification limits. If the computer is not working, there are backup procedures to provide the same information. The process computer has no specific regulatory requirements.

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Q. 031.026
(7)

Describe the installation, operation, and removal of the "Startrec" computer system which is used for startup testing of GE boiling water reactors, including the following topics: (a) specifications and qualification testing of electrical isolators; and (b) separation criteria for permanent and temporary wiring.

Response:

The Transient Data Acquisition System (TDAS) to support startup transient testing will no longer be the GE STARTREC computer system. A re-evaluation of the WNP 2 data acquisition needs has led to the implementaton of a permanent installation.

The WNP-2 TDAS control unit and analysis computer will be located in the main control room. Analog and digital signals will be isolated and conditioned in divisionalized remote units in the cabinet where they originate. Multiplexing and digitizing of all data will take place in these remote units before the data is transmitted over fiber-optic links to the control unit.

All signals originating from safety-related divisionalized equipment will be physically and electrically isolated such that faults occuring in the TDAS equipment cannot propagate back into safety-related circuits. The isolation devices will be qualified to the standards of Class 1E equipment and meet the intent of Regulatory Guide 1.75 concerning isolation devices

All TDAS input circuits within raceways are identified and routed to Class 1E requirements up to a remote isolation device. From the isolation device to the remote multiplexer the circuits are considered to be non-Class 1E.

Remote multiplexer outputs are transmitted to the computer via a fiber optic cable which is inherently an isolation device. The fiber optic cable, therefore, can be routed in any raceway without regard to separation criteria.

TDAS Class 1E input isolators are supplied from non-Class 1E 24 VDC current limiting power supplies. The power source to these power supplied is Class 1E and is provided with a Class 1E current interrupting device. The circuit to the power supply is routed as prime (see Section 8.3.1.4) for Division

1 and 2 isolators and as Class 1E for the Divsion 3 isolated from the Class 1E signal input circuit. Downstream of the power supply, the circuits are treated as non-Class 1E.

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Q. 031.027
(7.3.1.1)

The description of low pressure interlocks in 7.3.1.1.1.5 of the FSAR is incomplete. Describe the parameter sensed and the function of MO E12-F087, MO E12-F052, and MO E12-F051.

Response:

Section 7.3.1.1.1.7 has been revised to include:

RHR System	Type	Valve	Parameter Sensed	Function
Steam Condensing Mode	MO	E12-F087	Steam pressure	Provide low-pressure supplementary flow
	MO	E12-F052	None	Block valve
	AO	E12-F051	Steam pressure	Maintain system pressure

Amendment 10, revising Chapter 7, put all discussion of High Pressure/Low Pressure System Interlocks in 7.6.1.2.

Q. 031.028 (RSP)
(7.3.1.1)
(7.3.2.1)

The discussion of environmental conditions such as that in 7.3.1.1 and 7.3.2.1 is unacceptable. It is the staff's position that all safety-related equipment, including cables, must be qualified for operation in the worst case environment. Inside the containment, this design basis environment is established by postulated accidents. Equipment outside of containment must be qualified to the extremes of expected conditions which could result from the failure of other engineered safety features or equipment required to maintain a controlled environment such as plant heating systems. Accordingly, we require you to demonstrate compliance with this staff position. Identify and justify all exceptions.

Response:

All safety-related components including cables which are supplied by GE whether located in the containment or outside the containment are selected to meet the environmental conditions in 3.11. There are no exceptions taken.

See also revised 7.3.1.1.5.3, 7.3.1.1.6.3, 7.3.1.1.7.3, 7.3.1.1.10.3 and 7.3.1.1.11.3.

Qualifications of balance of plant cables and components is covered in 3.11 and 8.3.1.2.3.

Amendment 10, revising Chapter 7, replaced 7.3.1.1.5.3, 7.3.1.1.6.3, 7.3.1.1.7.3, 7.3.1.1.10.3 and 7.3.1.1.11.3 with 7.3.1.2.f. WNP-2 has a seismic and environmental qualification program which will modify 3.11.

Q. 031.029
(7.3.1.1)

Identify and justify all containment isolation valves which are provided with manual override of the isolation logic. Identify and justify all other aspects of the WNP-2 design which do not meet the requirements of 4.16 of IEEE Std. 279-1971. For each manual override which is provided in the WNP-2 design, demonstrate compliance with IEEE Std. 279-1971, 4.11 through 4.14.

RESPONSE:

There are no manual override on the isolation logic for the containment isolation valves. The protection systems for the primary containment and reactor vessel isolation control systems meet 4.16 of IEEE Std. 279-1971. Once initiated at the system level, the isolation protective action goes to completion. Return to normal operation requires deliberate operator action.

Q. 031.030
(7.3)
(7.6)

The design basis information such as that provided in 7.3.2.1.2 of the FSAR is unacceptable. Provide the following types of information in the appropriate sections of the FSAR:

a. Quantify the voltage and frequency (or pressure) level at which the power supply is defined to have failed.

b. Provide the range of temperatures and humidity over which the instrumentation and controls will meet their design bases (i.e., their environmental qualifications).

c. Include a discussion of the consequences to electrical equipment for a flooded sump and the consequences of a probable maximum flood.

d. Provide the response times and accuracies.

e. Describe the design provisions which prevent saturation and fold over.

RESPONSE:

a. The voltage and frequency level at which the power supply is defined to have failed are as follows:

Item	Qualified Limit: Input Voltage	Qualified Limit: Frequency
24V d-c power supply	130-V RMS max 97-V RMS min	± 5 Hz
120-V a-c/125-V d-c inverter	140-V d-c max 105-V d-c min	N/A

b. The standard temperature for the control room instrumentation is 40-120-F and 90% relative humidity (maximum). The range of temperatures and humidity over which the LPCI, LPCS, HPCS, ADS, and MSIV-LCS instrumentation and controls will meet their design basis is provided in Table 3.11-1, 3.11-2 and Ref 3.11-2.

c. Probable maximum floods have no (external) effect on Class 1E systems (see 3.4). The results of a detailed (internal) flooding analysis are summarized in the "Environmental Qualification Report for Safety-Related Equipment" referenced in Section 3.11. That summary includes: WNP-2 can be safely shut down with alternate safety-related equipment not affected by flooding and that flooded equipment will not defeat accomplishment of the safety function. Based on this, flooding is not a required qualification parameter for WNP-2 safety-related electrical equipment in secondary containment.

   The operator is alerted to flooding by leak detection sensors and can take action to initiate alternate safety systems not affected by flooding if required. As the operator is alerted to flooding, instrumentation that may subsequently become flooded and provide erroneous indication will not mislead the operator to take inappropriate action.

d. Instrument response times used in the WNP-2 simulation in Chapter 15 will be provided in the response to Question 0341.016. Table 7.2-1, 7.3-1, 7.3-3, 7.3-5, 7.3-7, 7.3-9, and 7.4-1 will be revised to include instrument accuracies in response to Question 031.021.

e. The differential pressure sensors (level switches and ΔP transmitters) are designed for one side pressurization capability of up to 2,000 psig without damage to diaphragm bellows.

Amendment 10, revising Chapter 7, replaced 7.3.2.1.2.3.1.19, 7.3.1.1.1.4.5, 7.3.1.2.3.1.20 with 7.3.2.1.2.a.19, 7.3.1.1.1.2, 7.3.2.1.2.a.20 and 7.3.1.1.1.2, respectively. Tables 7.3-2, 7.3-3, 7.3-4 and 7.3-5 were renumbered 7.3-3, 7.3-5, 7.3-7 and 7.3-9, respectively.

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Q. 031.031
(7.3.2.1)

The description of the automatic depressurization system with respect to the requirements of IEEE Std. 279-1971, 4.19 and 4.20, is inadequate. Provide the following information:

a. Describe how the operator is made aware of items (a) through (d) under the discussion of compliance with 4.19 of IEEE Std. 279-1971.

b. Provide justification for the use of the relief valve discharge pipe monitors and plant annunciators for providing information which forms the basis for operator action.

c. Define the term "ADS level".

Response:

a. Identification of Protective Actions is discussed in 7.3.2.1.2.3.1.19.

b. The ADS is a backup system to HPCS. Its function is to depressurize the reactor automatically in the event of a LOCA if the HPCS system fails to maintain vessel water level. Depressurization allows the low-pressure ECCS to do their job. No operator action is required. However, if, based on water level indications in the control room, the operator determines that the HPCS can restore water level without the aid of low-pressure systems, a reset switch is available to delay ADS initiator for 120 sec. The operator decision to delay ADS is made by looking at vessel water level indication and HPCS flow and pressure indication provided in the control room. See 7.3.1.1.1.4.5.

c. "ADS levels" refers to vessel water level indications which are applicable permissives for ADS initiation. That is, ADS initiates on vessel low water level Trip 1 and Trip 3. See 7.3.2.1.2.3.1.20.

Amendment 10, revising Chapter 7, replaced 7.3.2.1.2.3.1.19, 7.3.1.1.1.4.5 and 7.3.2.1.2.3.20 with 7.3.2.1.2(19), 7.3.1.1.1 and 7.3.2.1.2(20), respectively.

Q. 031.032 (RSP)
(7.2.1.1)
(7.3.2.1)
(15.2.6)
(F7.2-1)
(T15.2-13)

It is the staff's position that the use of Class 1E power supplies as alternate feeds for the reactor protection system buses as described in 7.2.1.1.2 of the FSAR is unacceptable since this prevents the required separation from the third division at level switch 1B21-NO24A and NO24C (GE Drawing 828E479TU). Accordingly, we require you to provide a revised design for the alternate power source for the reactor protection system which satisfies your electrical separation criteria. Identify and justify all exceptions. Clarify the discrepancy between the present design and the assumptions of 15.2.6 and Table 15.2-13.

RESPONSE:

Drawing 828E479TU is not applicable to WNP-2. See Figure 8.3-2.

Q. 031.033 (RSP)
(7.2.2.2)
(7.3.2.1)

The discussions of conformance with Regulatory Guide 1.47 do not indicate that the individual system level indicators can be actuated from the control room by the operator. It also appears that you take exception to: (a) the staff's position that administrative controls are not an acceptable substitute for the indication which is required by 4.13 of IEEE Std. 279-1971; and (b) that the design of bypass circuits for engineered safety features (ESF) shall include control room indication whenever operator actions result in the loss of an ESF function or a reduction in system redundancy. Accordingly, we require you to provide a revised design and describe the provisions which have been made for the manual initiation of the system level status alarms for those bypasses which are expected to occur less frequently than once per year.

RESPONSE:

(a) Conformance to 4.13 of IEEE 279-1971:

(1) For RPS:

Bypasses are indicated by annunciators.

(2) For RHR, HPCS, LPCS, RCIC:

Bypasses are associated with testing, maintenance, calibration, and inoperable conditions. All manual test switches are key-lock type so that administrative procedures must be in effect at the time of the test. Annunciation is provided to indicate test status.

(3) For RCIC turbine:

The RCIC turbine trip is annunciated in the control room, and the annunciation requires manual reset.

(4) For HPCS diesel generator:

The HPCS diesel generator 86 trips and the diesel engine 86 trips are annunciated in the control room, and the annunciation requires local, manual reset.

Q. 031.034
(7.3.2.1)

The description of compliance with the requirements of 4.19 and 4.20 of IEEE Std. 279-1971, is incomplete and, therefore, unacceptable. Provide the following additional information:

a. On a control room layout drawing, indicate the location of the relays and the limits of the area in which the operator is "at the controls" (as defined by Regulatory Guide 1.114) under non-emergency conditions.

b. Describe the testing of the annunciators and relay lights which qualify them for Class 1E service.

RESPONSE:

a. The description of compliance with IEEE Std. 279-1971 4.19 and 4.20 refer to trip actuation relays located in the control room, for indication and verification of trip occurrence.

The actuation relays for the following systems are located in the associated panels within the control room, and are observable without opening the cabinet doors.

RPS	- H11-P609 & P611
HPCS	- H11-P625
LPCI	- H11-P618 & P629
LPCS	- H11-P629
ADS	- H11-P628 & P631

These panels are located within the control room and accessible to the operator under non-emergency conditions.

The operator in the main control room is "at the controls" under non-emergency conditions. He has an unobstructed view of the reactor control benchboard, the reactor recirculation and reactor water cleanup benchboard, the reactor core cooling benchboard, and the balance of plant benchboard.

Regulatory Guide 1.114 does not have to be met for WPPSS-NP-NO. 2.

b. The function of the annunciators and relay lights is to supply information to the operator. They are not protective systems in that they do not provide any trip signals. No protective action is required based on the information supplied by the annunciators or relay lights. They are therefore not required to be qualified as Class 1E.

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Q. 031.035
(7.3.2.1)

Describe the design features which provide assurance that the main steam line isolation valves do not close in less than 3 seconds.

Response:

The MSIV does not close in less than 3 seconds due to an inline hydraulic damper restricting the closure speed. A small vernier type control limiting the hydraulic flow from below the damper piston allows accurate speed setting of valve closure. This speed control valve (#7 of Figure 7.3-4), having the capability to vary the MSIV closure speed from 3 to 10 seconds, can be set and locked for the desired speed. The control valve is set during production testing. MSIV closure speed is redetermined at preoperational tests and plant shutdown surveillance tests and control valves reset as required.

Amendment 10, revising Chapter 7, renumbered Figure 7.3-4 to 10.3-8.

Q. 031.036
(7.3.2.1)
(F7.3-8b)

Provide a drawing for the test control logic which shows how the main steam line isolation valve is tested for the required response time limit (e.g., greater than 2 seconds but less than 5 seconds) at rated steam flow.

Response:

Figure 7.3-11 shows the test control logic for testing the MSIV closure time. In order to avoid scram during MSIV full closure testing, the reactor power is reduced to approximately 75%.

The control room operator may time the valve closure while observing valve status lamps.

Amendment 10, revising Chapter 7, renumbered Figure 7.3-11 to 7.3-10.

Q. 031.037
(7.3.2.1)

The statement that "All components used in the isolation system have demonstrated reliable operation in similar nuclear power plant protection system or industrial application," is unacceptable since: (a) this statement does not satisfy the requirements of IEEE Std. 323-1971; and (b) considerable problems have been experienced with sensor drift. Provide an amended discussion of compliance with the requirements of 4.4 of IEEE Std. 279-1971 which satisfies the requirements of IEEE Std. 323-1971 and which describes the methods used to reduce sensor drift to acceptable levels.

Response:

Components were chosen on the basis of being the best available, vendor test and specs, and proven operational use in similar application. Operational experience has indicated some sensor drift, however, the frequency of surveillance check, test, and calibration and use of historical instrument data has consistently kept sensors within the limits of safety operation.

Conformance to IEEE Std. 323-1971 is discussed in 7.1.2.5.4.

Sensor drift is discussed in 7.1.2.4.

Amendment 10, revising Chapter 7, replaced 7.1.2.5.4 and 7.1.2.4 with 7.1.2.3.c and 7.1.2.5 respectively.

Q. 031.038
(7.3.1.1)
(7.3.2.1)

Clarify the description of the temperatures monitoring circuits which initiate containment and reactor vessel isolation. It is the staff's understanding that the BWR/5 and BWR/6 plants are equipped with a system using thermocouples. Include the following information in this clarification:

a. Describe the monitor system in the manner requested in 7.3 of the Standard Format.

b. Describe how the system satisfies the requirements of IEEE Std. 338-1971 and General Design Criterion 21 and how it conforms to the guidance in Regulatory Guide 1.22.

c. Clarify the discrepancy between 7.3.1.1.2.4.1.12.2.1 and 7.3.2.1.2.3.1.5.2.1.10 of the FSAR.

Response:

a. Main steamline space temperatures and differential temperatures are measured by dual-element thermocouples, and the analog signals are transmitted from the sensors to the temperature switch point modules located in the control room. This temperature measurement circuit arrangement is similar to Zimmer but not similar to Duane Arnold. Duane Arnold uses bimetallic temperature switches which are locally mounted. See 7.3.1.1.2.4.1.3.

b. The main steamline space temperature detection system, can be tested during reactor operation. Operability of the sensors (thermocouples), can be verified by comparing the readings during reactor operation. A complete check of the system, including the sensors, can be made during refueling and other planned shutdown periods.

c. Responses to items (a) and (b) help to clarify any conceived discrepancies.

See 7.3.2.2.2.3.1.9, 7.3.2.2.2.3.1.10, 7.3.2.2.2.3.4 and 7.3.2.3.2.2.1.

Amendment 10, revising Chapter 7, replaced 7.3.1.1.2.4.1.3, 7.3.2.2.2.3.1.9, 7.3.2.2.2.3.1.10, 7.3.2.2.2.3.4 and 7.3.2.3.2.2.1 with 7.3.1.1.2.b.4, 7.3.2.1.2.a.9, 7.3.2.1.2.a.10, 7.3.2.1.3.a and 7.3.2.1.1 respectively.

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Q. 031.039 (RSP)
(7.3.2.2)

The methods proposed for testing of some safety functions are unacceptable. Accordingly, we require you to provide modified designs for all of the safety-related instrumentation and control systems so that these designs will satisfy the following staff positions. Identify and justify any exceptions.

a. All portions of the protection systems shall be designed in accordance with IEEE Std. 279-1971, as required by 10CFR50.55a(h). All actuated equipment that is not tested during reactor operation, should be identified and a discussion of how this equipment conforms to the guidance contained in paragraph D.4 of Regulatory Guide 1.22, should be submitted.

b. The use of jury-rigged bypasses such as temporary jumpers, the removal of fuses, or removal of connectors is not an acceptable method for standard in-service testing.

c. The containment isolation valves for the WNP-2 facility shall be tested from the sensors through the actuating circuits and the valves themselves.

d. The methods which are provided for the testing of protection systems shall satisfy the requirements of 4.11 of IEEE Std. 279-1971.

RESPONSE:

a. Relative to IEEE 279-1971 and Regulatory Guide 1.22, which requires that actuated equipment be tested during reactor operation, all of the actuated equipment has the capability to be tested during reactor operation.

b. In no instance will it be necessary during testing of these circuits to either lift leads or remove fuses.

c. All isolation valves can be tested from the sensor through the actuating circuits.

d. See also revised 7.2.2.1.2.3.1.11, 7.3.1.1.5.3, 7.3.1.1.6.3, 7.3.1.1.7.3, 7.3.1.1.10.3, 7.3.1.1.11.3 and 7.3.2.2.2.3.1.11.

Amendment 10, revising Chapter 7, replaced the referenced paragraphs in d. above with 7.2.2.2.a.11, 7.3.1.2.f (for 7.3.1.1.5.3, 7.3.1.1.6.3, 7.3.1.1.7.3, and 7.3.1.1.11.3), and 7.3.2.1.2.a.11 respectively.

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Q. 031.040
(7.3.2.1)
(7.4.2.2)

Provide justification for your position stated in 7.3.2.1.2.3.4 and 7.4.2.2.2.2.3 of the FSAR that only 2.1 and 2.2 of IEEE Std. 338-1971 are applicable to the design of the emergency core cooling system and the reactor core isolation-cooling system. Identify and justify all exceptions to IEEE Std. 338-1971.

Response:

The statements in the subject paragraphs are in error and have been revised to state that the ECCS and RCIC comply fully with the requirements of IEEE 338-1971.

Amendment 10, revising Chapter 7, replaced 7.3.2.1.2.3.4 and 7.4.2.2.2.2.3 with 7.3.2.1.2 and 7.4.2.2 respectively.

Q. 031.041
(7.3.2.4)
(7.4)

The discussion in 7.3.2.4.4 and 7.4 of the conformance of the WNP-2 design with the present Regulatory Guides, is incomplete. Revise these sections of the FSAR to include a discussion of how the WNP-2 design conforms with Regulatory Guide 1.29.

RESPONSE:

NRC Regulatory Guide 1.29 has been addressed in revised sections of the FSAR. See revised 7.1.2.6.6.

Amendment 10, revising Chapter 7, replaced 7.1.2.6.6 with 7.1.2.4.b.

Q. 031.042
(F7.3-8b)

Revise all FSAR figures (such as Figure 7.3-8b) to include alpha-numeric area locators if such figures are referenced by, or continued on, additional sheets or figures.

RESPONSE:

FSAR figures have been revised to include alpha-numeric area locators.

Q. 031.043 (RSP)
(7.3.2.5)
(7.4.1.1)

It is the staff's position that trip devices such as "86" devices which require manual reset (i.e., lockout relays with manual reset), must have the tripped condition indicated on the inoperable and bypassed status indicator. Accordingly, we require you to provide a revised design for the diesel generator lock-out relaying, the reactor core isolation cooling system turbine over-speed trip, and any other similar device which is a part of a safety-related system. Identify and justify each such system which does not have all lock-outs indicated in conformance with the staff's position in Regulatory Guide 1.47.

RESPONSE:

Regulatory Guide 1.47 addresses functions which are "bypassed or made inoperable during the performance of periodic tests or maintenance." This guide describes the criteria for monitoring such activities. The devices noted in Question 031.043 are equipment malfunction trips. Trips of these devices do not occur as a result of operator or administrative action. Reset relays such as "86" relays are alarmed for equipment malfunctions, and equipment trip conditions are noted by amber lights on the main control board.

Q. 031.044
(7.4.2.2)

The discussion of compliance with the requirements of 4.17 of IEEE Std. 279-1971 is inadequate since it does not address manual initiation at the system level. Provide a description of manual initiation at the system level for all subsystems of the emergency core cooling system. Identify and justify all exceptions.

RESPONSE:

Manual Initiation (IEEE 279-1971, paragraph 4.17)

HPCS: The HPCS has an armed manual initiation pushbutton in parallel with the automatic initiation logic.

ADS: The ADS has four manual initiation switches. Two switches are in each of the two ADS systems (A&B). Both switches for one system have to be closed to manually initiate ADS. To further preclude inadvertent actuation, each switch is equipped with a collar which must be turned before electrical contacts of the pushbutton are effective. Thus, to initiate ADS manually, the operator must turn two collars and depress two pushbuttons. Whenever a collar is turned, an annunciator is actuated. The two switches have, as a permissive, the ac interlocks as described in 7.3.1.1.1.4.5.

The ADS automatic initiation delay timer is provided to give HPCS ample time to automatically restore vessel level so that ADS actuation will not be needed. This delay timer is not provided for manual initiation since the operator will not initiate ADS until he determines it necessary without further delay.

LPCS: The LPCS has an armed manual initiation pushbutton in parallel with the automatic initiation logic. This manual initiation will also initiate LPCI A.

LPCI: In no event can failure of an automatic control circuit for equipment in one division disable the manual electrical control circuit for the other LPCI division. Single electrical failures cannot disable manual electrical control for the LPCI function. LPCI A has an armed manual initiation pushbutton in parallel with the automatic initiation logic which will also initiate LPCS. The LPCI B and C systems have an armed manual initiation pushbutton in parallel with the automatic initiation logic.

Refer to 7.3.2.1.2.3.1.17 for the above description.

Amendment 10, revising Chapter 7, replaced 7.3.1.1.1.4.5 and 7.3.2.1.2.3.1.17 with 7.3.1.1.1.2 and 7.3.2.1.2.a.17 respectively.

Q. 031.045
(7.4.1.2)

It is the staff's understanding that the heating system for the standby liquid control system has been redesigned and now provides redundant heaters which are powered from Class 1E busses. Accordingly, we request that you: (a) revise 7.4.1.2.2; and (b) provide process and instrumentation drawings and electrical schematics which include the revised heating system and its controls.

Response:

a. The standby liquid control system heaters have not been redesigned. The system remains with two heaters, one of which is used for initial heating (when rapid heating is required), the other is used to maintain the SLCS solution at required temperature. The heaters and their controls are not required for the initiation of the SLCS.

   Section 7.4.1.2.2 is correct and requires no updating.

b. There are no revised heating system drawings for the SLCS.

   See Figures 7.4-3 and 7.4-4.

Amendment 10, revising Chapter 7, replaced 7.4.1.2.2 with 7.4.1.2.

Q. 031.046
(7.4.2.4)

You indicate in 7.4.2.4.2 of the FSAR that the only Regulatory requirements applicable to the design of the residual heat removal (RHR) system are General Design Criteria (GDC) 34 and 61. However, GDC 34 states, in part, that the residual heat removal system has a safety function and requires that "... suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities" be provided. Accordingly, revise your design to provide a leak detection capability and provide the appropriate information to demonstrate that the RHR system satisfies all the requirements of the General Design Criteria.

Response:

Section 3.1.2.4.5 contains descriptive material which provides a discussion of RHR system compliance with General Design Criterion 34. Section 5.2.5 contains a discussion of the Leak Detection System and its application to the RHR System. Section 15.2.9 discusses a backup method for disposing of residual heat should the normal shutdown line become unavailable during shutdown.

Amendment 10, revising Chapter 7, replaced 7.4.2.4.2 with 7.4.2.1.

Q 31.047
(7.5.2.4)

The seismic qualification of indicators and recorders for post-accident monitoring which is described in 7.5.2.4 is unacceptable. It is the staff's position that post-accident indicators and recorders must meet their minimum performance requirements before and after a seismic event without requiring adjustments or repair. (The staff acknowledges that these electro-mechanical devices may not provide accurate readings during severe vibrational excitation). Accordingly, we require you to provide a revised design which satisfies the staff's position on this matter.

Response:

The indicators and recorders are nonseismic. At the time WPPSS-2 was being designed, there were no IEEE standards or Regulatory Guide requirements for design of the subject instrumentation.

The instrumentation and readout devices are of high quality and from well-known manufacturers. This similar instrumentation is used for post-accident monitoring in such licensed and operating plants as Duane Arnold and Brunswick 2, and in such plants as Zimmer and LaSalle presently in the late stages of review. Therefore, it is the position of the General Electric Company that the instrumentation provided is adequate.

Amendment 10, revising Chapter 7, replaced 7.5.2.4 with 7.5.2.

Q. 031.048
(7.4.1.4)

Identify the systems and functions controlled by the transfer switches referenced in 7.4.1.4.3e of the FSAR. Indicate the location of these switches.

RESPONSE:

The transfer switches are located on panel C61-P001 (remote shutdown panel) which is located outside the main control room. Selection of the location is based upon having no effect on the panel from the control room evacuation event. See also revised 7.4.1.4.6.

Amendment 10, revising Chapter 7, combined 7.4.1.4.3.e and 7.4.1.4.4.6 into 7.4.1.4.

The systems and functions controlled by the transfer switches are identified in FSAR section 7.4.1.4.

Q. 031.049
(7.6.1.5)
(F7.6-16)
(15)

The post-treatment monitors of the air ejector off-gas radiation monitoring systems are required in order to isolate the off-gas systems outlet and drain valves. They are assumed to function in the accident analyses presented in Chapter 15. The staff's review of the off-gas radiation monitoring post-treatment subsystem indicates that the design does not meet the single failure criterion and is, therefore, unacceptable. Provide a discussion of how the design will prevent the formation of water seals in the sample lines as a result of condensation of the sample gas or by flooding caused by the lightning protection spray.

RESPONSE:

The Standard Review Plan no longer requires this analysis. In the original analysis provided for WNP-2 no credit was taken for post-treatment monitor operation. The analysis assumed complete failure of the gas treatment system component with the highest fission product inventory with direct release of a specific fraction of the inventory directly to the environment. For the purpose of this analysis, isolation of the off-gas system is not required and failure of the air reactor off-gas radiation manufacturing was assumed. With these assumptions, there is no protection provided by the post-treatment monitors or the gas treatment system isolation valve which the post-treatment monitor would signal to close if high radiation levels are detected. The single failure criterion, therefore, is not applicable to the post-treatment monitor.

However, it should be noted that the post-treatment monitor provides an isolation valve closure signal if the two channels are in any combination of inoperative, downscale and high-high radiation trips. Power loss to the post-treatment monitor also initiates isolation valve closure.

Q. 031.050
(7.6.1.3)

The design provisions for the detection of leakage from the emergency core cooling systems (ECCS) during the long-term recovery following a postulated loss-of-coolant accident, are not clearly indicated. Accordingly, provide the following information:

a. Identify the Class 1E instruments which are available to detect leakage from the ECCS during the long-term recovery period cited above. (The fluid temperatures will be below the alarm set points for the area temperature detectors at this time).

b. For each instrument identified in response to item (a) above, indicate the range and accuracy and describe the indicator location and the type of indication provided.

c. Provide the range and accuracy of the suppression pool level instrumentation.

d. Provide a table which compares each of the accuracies of items (b) and (c) above to a corresponding rate of change in suppression pool level.

RESPONSE:

a. Detection of leakage from ECCS systems during the long term post-LOCA cooldown recovery will involve leak detection from RHR system. RHR room ambient and differential temperatures can be monitored from the control room. RHR flow and reactor water level can be also monitored from the Control Room to detect a large leak in the RHR system. All of these instruments are Class 1E instruments.

Although it is not necessary for other ECCS systems to be functioning during the long-term recovery period following a LOCA, indication is available on each of the other systems which could provide indication of ECCS leakage. These indicators consist of flow measuring instrumentation on the HPCS and LPCS.

b. Range, accuracy, type and location indication of these Class 1E instruments are:

Instrument	Range	Accuracy	Indicator Type and Location
RHR Rm. A Amb. Temp.	50-350°F	±5°F	Common Meter, Control Room
RHR Rm. B Amb. Temp.	50-350°F	±5°F	Common Meter, Control Room
RHR Rm. A Diff. Temp.	0-150°F	±5°F	Common Meter, Control Room
RHR Rm. B Diff. Temp.	0-150°F	±5°F	Common Meter, Control Room
RHR Flow	0-10,000 gpm	±1.5%	Meter, Control Room
Reactor Water Level	-150/0/+50 in.	±1.5%	Meter, Control Room
HPCS Flow	0-8,000 gpm	±160gpm	Meter, Control Room
LPCS Flow	0-10,000gpm	±85gpm	Meter, Control Room

In addition, although it is not a Class 1E instrument, sump level also will give indication of leak.

c. The range and accuracy of the suppression pool level instrumentation is given in Table 7.3-1.

d. The range and accuracy of the suppression pool level instrumentation can not be meaningfully correlated with a "rate of change" of suppression pool water level.

Q. 031.051
(7.6)
(T7.1-1)

The descriptions of the systems and components presented in 7.6 of the FSAR are inadequate. Provide the information requested in 7.6 of the Standard Format, including a discussion of all differences between the designs of these systems and BWR-5 designs such as Zimmer and LaSalle.

RESPONSE:

The description of the systems and components present in 7.6 of the FSAR has been rewritten. Similarity to licensed reactors is illustrated by revised Table 7.1-2.

Q. 031.052 (RSP)
(7.1.2.1)
(7.6.2.6)
(15.4.2.2)
(T7.1-2a)

The accident analysis presented in 15.4.2.2 of the FSAR is based, in part, on the assumption that the rod block monitor (RBM) acts to mitigate the consequences of a continuous withdrawal of a control rod. Accordingly, it is the staff's position that the RBM is a protection system and must be designed, fabricated, installed, tested, and subjected to all of the criteria applicable to a reactor trip system. Accordingly, we require you to revise your design to reflect the importance of the RBM. Identify and justify any exceptions.

Response:

The RBM is used to prevent the operator from withdrawing a rod in the power range so that fuel cladding integrity is always maintained. It has two channels that are redundant, separated, and isolated. It provides two alarms and then a block as the power goes up locally.

The rod block monitor (RBM) is integral with the neutron monitoring system which is presently identified in 7.1.2.1.4. The RBM design is included in 7.1.2.1.4.6. The RBM is adequately described in terms of initiating circuits, logic, bypasses, interlocks, redundancy, diversity, actuated devices in 7.6.1.5.7. It is to be emphasized that the WNP-2 RBM is essentially identical to the RBM which has been licensed previously on all plants from Duane Arnold through Hatch-1 and Brunswick. The design of the RBM of WNP-2 is identical to the design of the RBM for Zimmer.

The Chapter 15 and Appendix 15A safety analysis involving the continuous rod withdrawal error, presents the event as an anticipated operational transient.

The RBM is not used in any Design Basis Accident (DBA) category event described in Chapter 15.

Amendment 10, revising Chapter 7, removed the discussion of the RBM from 7.1 and 7.6 and consolidated the discussion in 7.7.1.8.

BLANK

Q. 031.054
(7.7.1.11)
(8.3.1.1)

The material presented in 7.7.1.11 of the FSAR is inadequate. Provide the following additional information:

a. Clarify the discrepancy between the statement in 7.7.1.11 that the reactor water cleanup system is fed from the plant instrumentation bus and does not require back-up power and that in 8.3.1.1.3 which described these busses as Class 1E.

b. Since overpressure protection is a function of the reactor water cleanup system instrumentation, provide justification for not providing a Class 1E system.

c. Provide justification for not providing Class 1E equipment for the isolation functions listed in 7.7.1.11.1.1 (sic) of the FSAR.

Response:

a. There is no discrepancy existing between 7.7.1.8 and 8.3.1.1.3.

b. The RWCU system is not a safety-related system beyond the isolation valves. Overpressure protection for the system is provided by primary pressure relief valves. Therefore, there are no instrumentation requirements relative to overpressurization of the RWCU system.

c. Class 1E equipment is provided for system isolation. The RWCU system isolation valves are part of the reactor coolant pressure boundary (RCPB) and as such are controlled by the primary containment and reactor vessel isolation control system instrumentation. These valves and piping are Seismic Category I, and the controls are Class 1E as described in 7.3. The portion of the RWCU system outside the outer isolation valves is not part of the RCPB and not safety-related, and instrumentation for this portion is nonessential.

Amendment 10, revising Chapter 7, removed the subject discussion from Chapter 7 as adequate discussion is given in 8.3.1.1.3 and 5.4.8 of the RWCU system.

Q. 031.055
(3.10.1.1)
(031.001)
(031.044)

Identify all Class 1E equipment which was not qualified by test. For each such item, provide the basis for assuming that it will not be spuriously operated, or fail to operate when required, during and after a seismic event.

Response:

Class 1E equipment qualification documents whether qualified by test, analysis, or both is assembled in qualification files and has been summarized in "WNP-2 Dynamic Qualification Report for Safety-Related Equipment", dated September 1982. The qualification program includes consideration of the effects upon Class 1E equipment of seismic loads or seismic and hydrodynamic load combinations (including spurious operation). The program demonstrates adequacy of the methods and results of the Class 1E equipment qualifications as equal or conservative to the requirements of IEEE 344-1975.

Q. 031.056
(3.11)

Describe the environmental qualification procedures and the environmental extemes of qualification for the following specific passive Class 1E components inside the drywell: (1) splices; (2) terminal blocks; (3) termination cabinets; and (4) connectors.

Response:

See response to Question 031.006.

Q. 031.057
(3.11)

Identify all Class 1E equipment inside the drywell, except the equipment cited in Item 031.056, and summarize the environmental qualifications for this equipment. The identification and summary for each item should include: (1) the safety function and functional requirement; (2) the manufacturer, model number, type number, and any other identifying numbers; (3) the specific location of this equipment in the drywell; (4) the method of environmental qualification; (5) the environmental extremes, including the time period of testing, for which it is qualified; and (6) the identification and the location of the documents which are available so as to permit an independent evaluation of the adequacy of the environmental qualification.

Response:

See response to Question 031.006.

Q 031.058
(031.001)

Your response to Item 031.001(e) is incomplete. Describe the passive design features which prevent motion of the recirculation flow control valve after an accident. This motion could result from postulated damage to the hydraulic system during a loss-of-coolant accident.

Response:

As stated in the response to 031.001(e), the valves are mechanically disabled in the full open position.

The passive design features associated with the recirculation flow control valve to resist unaccountable motion in an accident are as follows:

1. Minimum exposure of components to the accident environment - The only components in the recirculation flow control valve system subject to the direct effects of a LOCA are the valve and valve actuator. The actuator is disabled mechanically to keep the valve in full open position.

2. Physical remoteness from potential pipe breaks -

   a) Referring to the response to 31.001(e), it is recalled the valve of concern is the one in the unbroken loop. Accordingly, for recirculation pipe breaks, there is virtually no potential for missile or jet impingement impact on the valve in the unbroken loop since it is on the opposite side of the reactor vessel within primary containment from the other loop.

   b) For other than recirculation pipe breaks, over 25 vertical feet of separation for the actuator exists from the nearest high energy line.

3. <u>Physical Protection from Effects of Pipe Breaks</u>

Substantial direct physical protection is offered as follows:

a) Valve actuator - The mechanically disabled valve actuator is located within the confines of large structural beams at the 512 foot elevation.

Accordingly, the risk is minimal for direct LOCA effects and the valve can be expected to function as described in the response to 031.001(e).

DELETED

Q. 031.059
(031.001)
(11.6-1)

Your response to Item 031.001(h) presents a new design for the logic of the main steam line isolation valves which is different from that reviewed and accepted for licensing on similar boiling water reactors. Provide the manufacturing drawings for ASCO Valve No. 832320. Additionally, provide the results of the engineering analysis and the test results which demonstrate the ASCO Valve No. 832320: (1) is qualified for the environment in the drywell following a loss-of-coolant accident; (2) is seismically qualified; (3) meets the physical separation and the required electrical independence in accordance with the staff positions contained Regulatory Guide 1.75; (4) satisfies the single failure criterion (previous designs accepted for licensing have used two separate valves in a one-out-of-two logic for a reactor trip). Note that Table 1.6-1 of the FSAR states that the GE Topical Report, APED-5750, is applicable to the WNP-2 facility and that Table 7.1-1 indicates the main steam line isolation valves are designed and supplied by GE. Accordingly, provide justification for the change to the design which was previously reviewed and approved by the staff in our evaluation of the GE Topical Report, APED-5750.

Response:

The main steam line isolation valve logic is the same for WNP-2 as that supplied for previously reviewed and accepted for licensing BWRs.

a. The equipment qualification reevaluation effort at WNP-2 determined that the solenoid pilot valves on the main steam isolation valve required replacement in order to provide units qualified for the environment potentially experienced. The original replacement valves were ASCO Model NP8323A20E. Environmental qualification information for these valves was included in the Supply System's Environmental Qualification Report referenced in Section 3.11. BDC 89-0356-0A subsequently replaced the ASCO dual-solenoid valve with Valcor single-solenoid valves. Therefore, the previous design accepted for licensing using two separate valves has now been installed. Environmental and seismic qualification of the Valcor valves was performed under the BDC.

b. The original solenoid valve was qualified for original seismic requirement when tested with complete valve (Wyle Laboratories -- Seismic Simulation Test Report #42610-1, dated 2/27/74). The solenoid valve remained functional during all phases of the testing. Reevaluation of the seismic qualification followed the decision to replace the original valve with Model NP8323A20E. The subsequent replacement valve (Valcor) was determined to be qualified and the qualification documentation is maintained in the Supply System qualification program files.

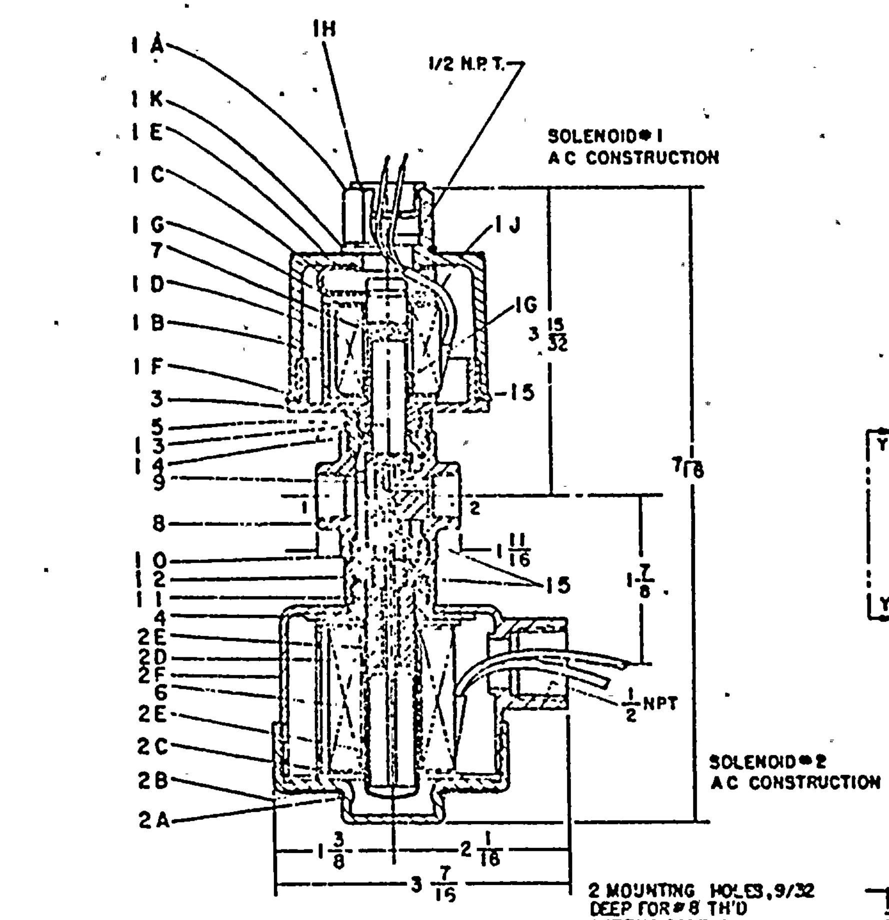
c. The protection system criteria of IEEE 279-1971 are met with this design; the requirements of Regulatory Guide 1.75 were not committed for this plant.

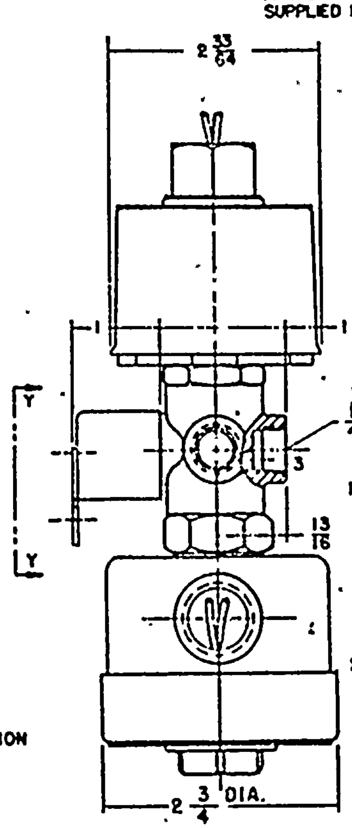
d. The Valcor valves in question are not used in generating a reactor trip. The Valcor valves are used in a two-out-of-two logic for each MSIV. That is, in order for each MSIV to be isolated both Valcor solenoids must deenergize. The Valcor valves themselves are not single failure proof. Single failure criterion is preserved since each main steam line contains two valves in series. If a single failure occurs in one valve scheme the second will provide isolation.

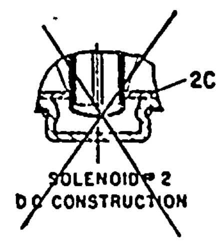
There is no deviation from the commitments made in APED-5750.

Because the decision to replace the valves has been made, the qualification concern is satisfied and the need for valve drawings is presumed to no longer exist.

GIVE FULL NAMEPLATE DATA, AND ASSEM. DWG. NO. WHEN ORDERING PARTS. ✱ INDICATES ITEMS SUPPLIED IN SPARE PARTS KITS.





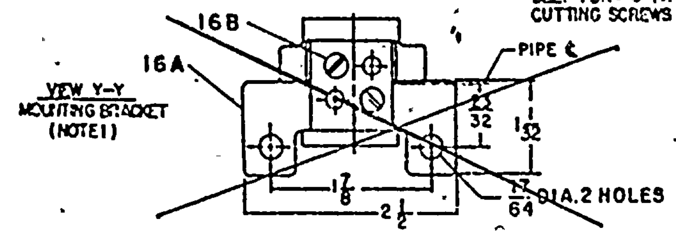


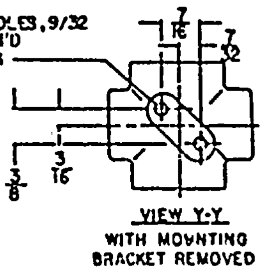
NOTES:

1- OPTIONAL MOUNTING BRACKET SUPPLIED ONLY WHEN SPECIFIED "WITH BRACKET". BRACKET MAY BE ROTATED TO ANY OF FOUR POSITIONS, IN INCREMENTS OF 90° FROM POSITION SHOWN.

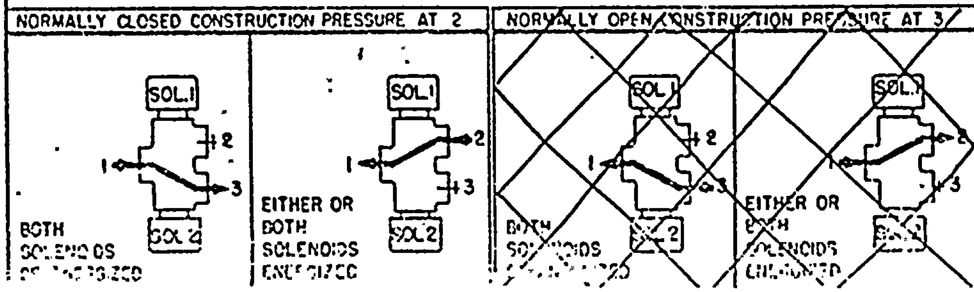
2- + USE CLASS 'B' HIGH POWER COIL FOR CATALOG NO'S. 832322, 832330, 832338 & 832346 ONLY. (SOL. #1)

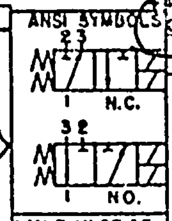
ITEM	NAME	QTY
1 A	HOUSING	1
1 B	YOKE	1
1 C	SPACER	1
1 D	COIL	1
1 E	NAMEPLATE	1
✱ 1 F	GASKET, HOUSING	1
1 G	SLEEVE	2
1 H	THREAD PROTECTOR	1
1 J	TUBING-INSULATING	1
1 K	SNAP RING	1
2 A	RETAINING RING	1
2 B	NAMEPLATE	1
2 C	YOKE	1
2 D	COIL	1
2 E	SLEEVE (A C)	2
2 F	COVER/HOUSING SUB-ASSEMBLY	1
3	SOL BASE SUB-ASSEM. (UPPER)	1
4	SOL. BASE SUB-ASSEM. (LOWER)	1
✱ 5	CORE ASSEM. (UPPER)	1
✱ 6	CORE ASSEM. (LOWER)	1
✱ 7	SPRING, CORE	1
8	BODY	1
✱ 9	DISC HOLDER ASSEM	1
✱ 1 0	SPRING, DISC	1
1 1	STEM	1
1 2	PLUGNUT ADAPTER ASSEMBLY	1
1 3	ADAPTER	1
✱ 1 4	GASKET, ADAPTER	1
✱ 1 5	GASKET, BODY	3
1 6A	BRACKET, MOUNTING	1
1 6B	SCREW, BRACKET	2





CATA. NO.	PIPE	ORF.	OPER.	WATTAGE SOL. #1	WATTAGE SOL. #2	SPK. NUMBER	NET WEIGHT
~~832348~~	1/4	~~1/8~~	NORM. OPEN	~~6 A C~~	~~10.8 D.C.~~	~~FV-166-263~~	3.1 LB.
+ ~~832346~~		~~3/32~~		~~9 A C~~		~~FV-166-262~~	
~~832344~~		~~1/16~~		~~6 A C~~			
~~832340~~		~~1/8~~	NORM. CLOSED			~~FV-166-263~~	
+ [illegible]		[illegible]		~~9 A C~~		~~FV-166-262~~	
[illegible]		[illegible]		~~6 A C~~			
[illegible]		~~1/8~~	NORM. OPEN		15.4 A.C.	~~FV-166-263~~	
+ [illegible]		[illegible]		~~9 A C~~		~~FV-166-262~~	
[illegible]		~~1/16~~		~~6 A C~~		~~FV-166-263~~	
[illegible]		~~1/8~~	NORM. CLOSED				
[illegible]		[illegible]		~~9 A C~~		FV-166-262	
832320		1/16		6 A C			





VALVE MUST BE MOUNTED WITH SOL. IN VERTICAL POSITION

BULLETIN 8323 3 WAY

DUAL SOLENOID VALVE

PIPE PORT CONN. 1/4 N.P.T. BODY MAT'L BRASS

ORIFICE DIA. (SEE TABLE) SEATING RESILIENT

ENCLOSURE WATERTIGHT/EXP. PROOF

WATTS (SEE TABLE)

HVA-166-265

SECTION 4
PAGE 12

INSTALLATION AND MAINTENANCE INSTRUCTIONS

GENERAL PURPOSE AND EXPLOSION PROOF WATERTIGHT SOLENOIDS

CATALOG
80161 80163
80162 80164

DESCRIPTION
Catalogs 80161 and 80162 solenoids are equipped with a General Purpose, NEMA Type 1 Enclosure. When Catalog 80161 or 80162 is installed just as a solenoid and not attached to an ASCO valve, the core has a 1/4-28-UNF-2B tapped hole, 3/8 full thread.
Catalogs 80163 and 80164 solenoids are equipped with an enclosure which is designed to meet NEMA Type 4 — Watertight, NEMA Type 7 (C or D) Hazardous Locations — Class 1, Group C or D and NEMA Type 9 (E, F or G) Hazardous Locations — Class II, Group E, F or G. When Catalog 80163 or 80164 is installed just as a solenoid and not attached to an ASCO valve, the core has a 1/4-28-UNF-2B tapped hole, 3/8 full thread.

OPERATION
When the solenoid is energized, the core pulls into the solenoid base sub-assembly. IMPORTANT: When the solenoid is de-energized the initial return force for the core whether developed by spring, pressure or weight must exert a minimum force to overcome residual magnetism created by the solenoid. Minimum return force for A-C Construction is 11 ounces and 5 ounces for D-C Construction.

INSTALLATION
Check nameplate for correct catalog number, voltage, wattage and service.

IMPORTANT: For the protection of the solenoid valve or operator install a strainer or filter suitable for the service involved in the inlet side as close to the valve or operator as possible. Periodic cleaning is required depending on service conditions. See Bulletins 8600, 8601 and 8602 for strainers.

WIRING
Wiring must comply with Local and National Electrical Codes. For Catalog Numbers 80163, 80164 electrical fittings must be approved for use in the approved hazardous locations. Housings for all solenoids are made with connections for 1/2 inch conduit. The general purpose enclosure may be rotated to facilitate wiring by removing the retaining cap. Rotate enclosure to desired position. Replace retaining cap before operating.

NOTE
Alternating Current (AC) and Direct Current (DC) solenoids are built differently. To convert from one to another, it is necessary to change the complete solenoid.

CAUTION
Core must be taken not to mar the core surface.

SOLENOID ENCLOSURE ASSEMBLY
Catalog 80161, 80162, 80163, and 80164 may be assembled as a complete unit. Tightening is accomplished by means of a hex flange at the base of the solenoid enclosure.

SOLENOID TEMPERATURE
Standard solenoids are supplied with coils designed for continuous duty service. When the solenoid is energized for a long period, the solenoid enclosure becomes hot and can be touched by hand only for an instant. This is a safe operating temperature. Any excessive heating will be indicated by the smoke and odor of burning coil insulation.

MAINTENANCE

CLEANING

A periodic cleaning of all solenoid valves and operators is desirable. The time between cleanings will vary, depending on the media and service conditions. In general, if the voltage to the coil is correct, sluggish valve operation, excessive leakage or noise will indicate that cleaning is required.

IMPROPER OPERATION

1. Check the electrical system by energizing the solenoid. A metallic click signifies the solenoid is operating. Absence of the click indicates loss of power supply. Check for loose or burned-out fuses, open-circuited or grounded coil, broken wires or splice connections. Replace coil if necessary.
2. Check voltage across the coil leads with a voltmeter. Voltage must be at least 85% of nameplate rating.

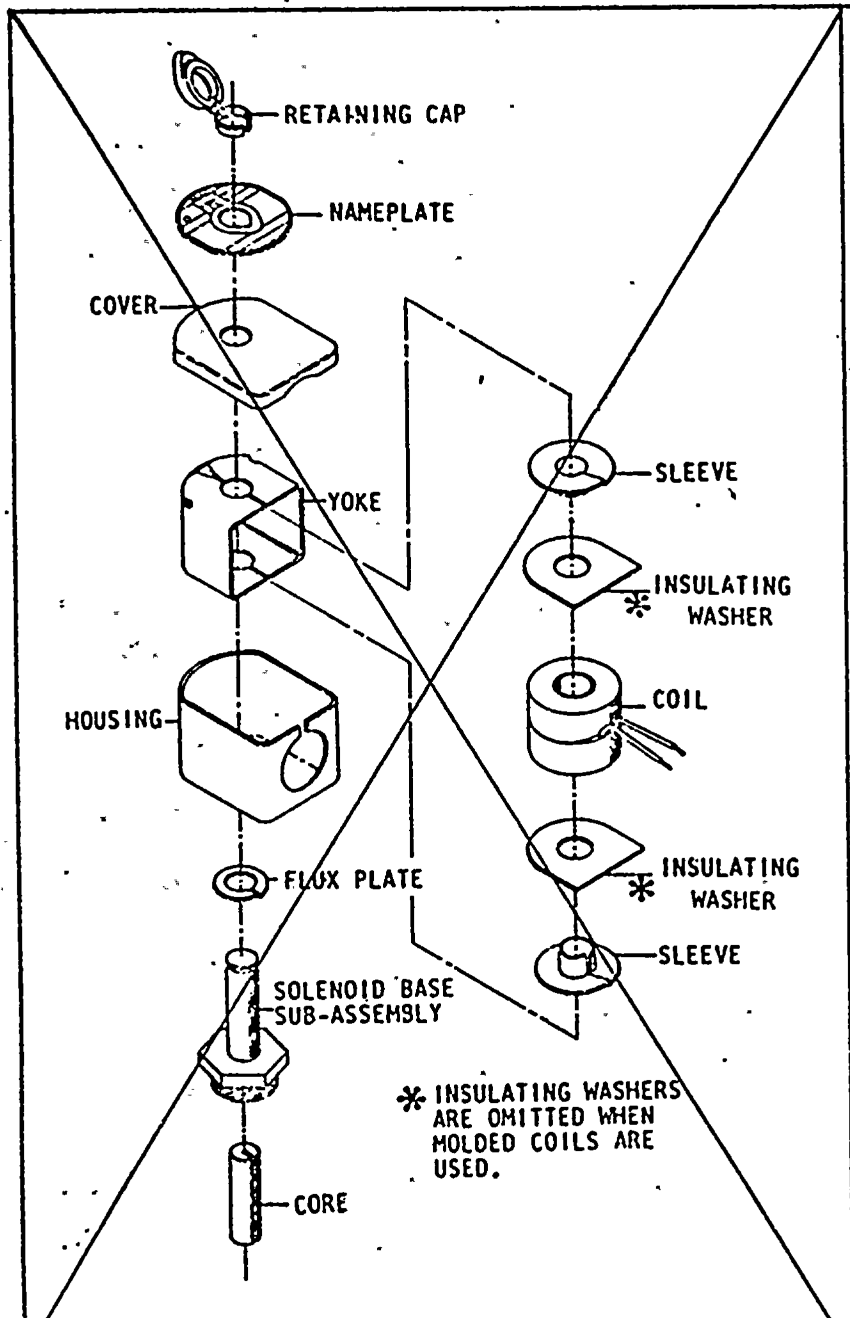
COIL REPLACEMENT

1. Turn off electrical power and disconnect coil lead wires.
2. Remove parts in the following manner: (Refer to exploded view on reverse side.
   A. Catalog 80161, 80162
      1. Remove retaining cap, nameplate and housing cover.
      2. Slip yoke containing coil, sleeves and insulating washers off the solenoid base sub-assembly. Insulating washers are omitted when a molded coil is used.
      3. Coil is now accessible for replacement. Reassemble in reverse order of disassembly.
   B. Catalog 80163, 80164
      1. Unscrew housing cover with retaining ring and nameplate attached.
      2. Remove spacer from the top of the yoke.
      3. Slip yoke containing coil, sleeves and insulating washers off the solenoid base sub-assembly. Insulating washers are omitted when a molded coil is used.
      4. Coil is now accessible for replacement. Reassemble in reverse order of disassembly.

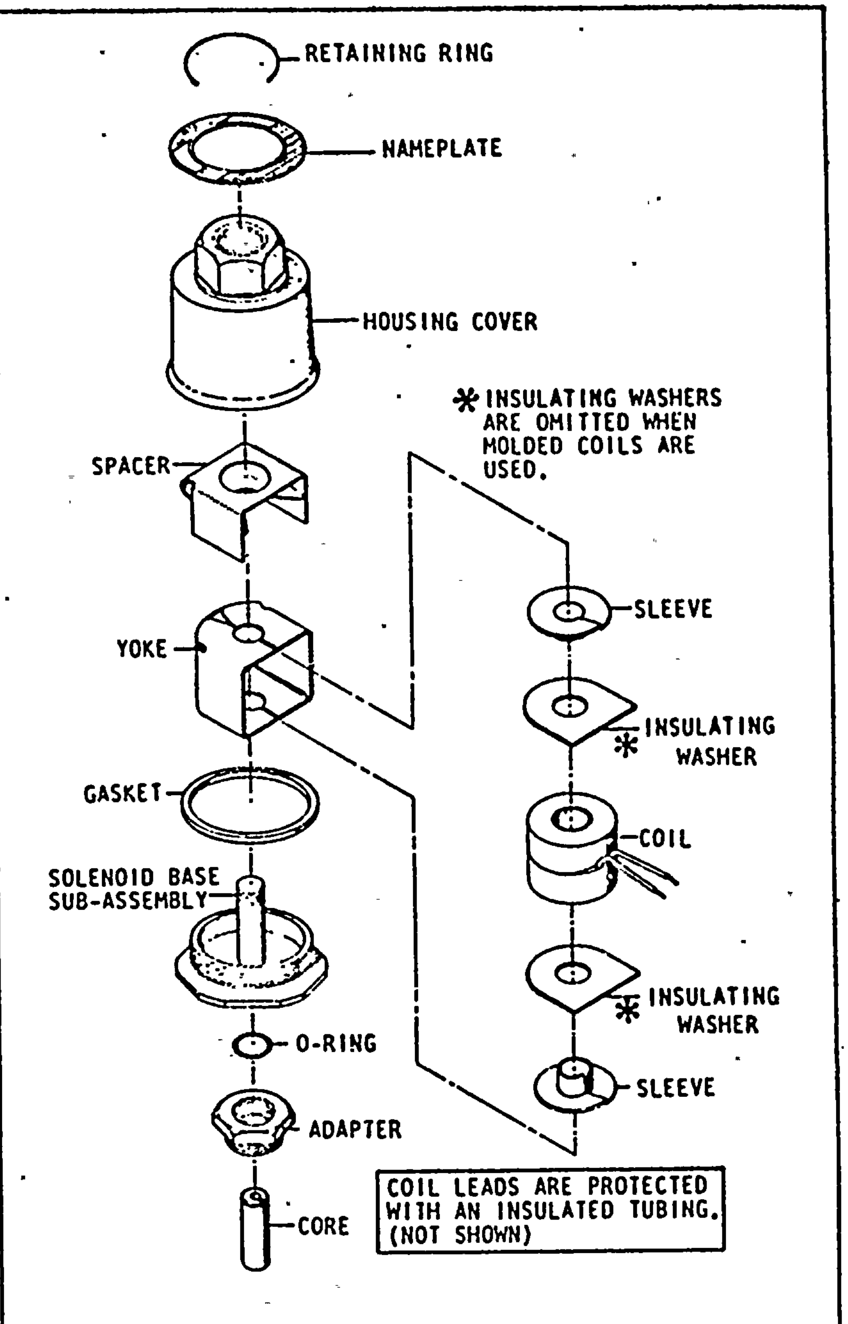
CAUTION
The solenoid must be fully reassembled as the housing and internal solenoid parts are part of and complete the magnetic circuit.

NOTE:
Catalog 80163, 80164
Installation and Maintenance of Explosion Proof equipment requires more than ordinary care to insure safe performance. All finished surfaces of the solenoid are constructed to provide a flameproof seal. Be sure that surfaces are wiped clean before replacing.

ORDERING INFORMATION FOR SPARE PARTS
FOR SPARE PARTS SPECIFY SERIAL NUMBER, CATALOG NUMBER, VOLTAGE AND INDIVIDUAL PARTS BY NAME.



CATALOG ~~80161, 80162~~



CATALOG 80163, 80164

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Q 31.060
(RSP)
(031.001)

Your response to Item 031.001(i) is unacceptable since it contradicts information contained in Reference 2 of your response. Specifically, the short term objectives presented in parts 1 and 2 of your response appear to preclude prompt cessation of feedwater flow which is stated in part 3 of the response to be desirable. Accordingly, we require you to provide a response which is consistent with the material which has been submitted by General Electric to the staff for our review of anticipated transient without scram.

Response:

An inconsistency does not exist because parts 1 and 2 of Question 31.001(i) should not be compared with part 3. The coast down of feedwater flow as in the ATWS studies mitigates the vessel pressure, containment temperature and pressure rises and, as such, is a benefit in the initial stages of the transient. By the time the feedwater lines are needed for makeup water, the transient (as far as ATWS is concerned) is over. Using the feedwater lines to supply makeup water from the condensate system pumps is considered for "short term" containment isolation but, in fact, occurs chronologically much later than in the ATWS studies.

Q 31.061
(031.039)

Your response to Item 031.039(b) is incomplete. Describe the design provisions which eliminate the use of jumpers other than test instrumentation leads.

Response:

Jumpers are not used except for the SCRAM initiation on SRM high flux trip during initial plant startup. This trip is jumpered out with permanent terminal block links after startup is completed.

Q 31.062

It is the staff's position that the responses to Items 031.001(ff), 031.002, and 031.023 do not provide the detailed design information which is required for an independent review of an application for an operating license. Accordingly, we require you to provide amended responses to the items cited above. Specifically, identify the bus requested in Item 031.001(ff), provide the information requested in Item 031.002(e) and Item 031.023(d).

Response: (to 31.001(ff)only)

The reactor manual control system and refueling interlocks are supplied at 120VAC via the 120/240 volt critical (Class 1E) instrumentation ac power system panels PP-7A and PP-8A, except for the fan control alternate source which is supplied via the 120/240 volt plant critical ac power system panel PP-8A-A. Details of supply are as follows:

Panel	Div.	G.E. Board	Via
PP-8A	2	H13-P603	PP-8A-Z, H12-P679
PP-8A-A	2	H13-P603	H13-P679
PP-7A	1	H13-P615	PP-7A-Z, H13-P688
PP-7A	1	H13-P616	H13-P688

Response: (to 31.002(e))

Drawings are provided in the response to question 31.068.

Response: (to 031.023(d))

See the response to question 31.055.

Q 31.063
(31.006)
(31.014)
(31.016)
(31.026)
(31.047)

Your responses to Items 31.006, 31.014, 31.016, 31.026, and 31.047, are unacceptable. Specifically, your responses to these particular items indicate that you will respond at a later date. Accordingly, provide a schedule for your response on these matters.

Response:

31.006 - See revised response.

31.014 - See revised response.

31.016 - Additional information with reference to question 31.006 is as follows:

The tables referenced in the responses to question 31.016 and 31.021 have been revised to indicate that the Technical Specifications for WNP-2 will contain instrumentation setpoint, response time, accuracy, and testing requirements necessary to insure minimum performance not less than those assumed in the safety analysis. Requirements listed currently in the tables are subject to change pending final approval of the Technical Specifications specific to WNP-2. The Technical Specifications and Instrumentation setpoints and bases for WNP-2 are based on NUREG 0123 Rev. 1, Standard Technical Specifications for General Electric Boiling Water Reactors, (April, 1978), and are in review at WPPSS having received the majority of the required input from General Electric. It is anticipated they will be ready for submission to the NRC in the second quarter of 1979.

31.026 - See revised response.

31.047 - See revised response.

Q 31.064
(31.009)

Your response to Item 31.009 is incomplete. Describe the consequences of a turbine trip without bypass which is coincident with a postulated failure of the electrical supply to actuate the "relief" function of the safety-relief valves. Describe all other loads which are powered from this particular electrical supply.

Response:

It is considered highly unlikely that a turbine trip without bypass transient with coincident failure of the electrical supply to actuate the "relief" function of the safety-relief valves would occur. The peak pressure during a turbine trip without bypass transient with coincident failure of the "relief" function of the safety-relief valves is less severe than the safety-relief valve sizing transient (i.e., MSIV closure with indirect scram) which only takes credit for the spring-action mode of operation of all safety-relief valves.) Furthermore, the peak fuel surface heat flux during a turbine trip without bypass occurs before the actuation of the safety-relief valves in their relief mode. Therefore, there is no effect on the MCPR limit during this transient if safety-relief valves fail to operate in their relief mode.

Q 31.065
(31.010)

Your response to Item 31.010 is incomplete. Identify the specific recirculation flow control valve components which would be damaged by the collapse of bubbles in the pressure recovery areas of these valves.

Reponse:

In response to 31.065, enclosed is a figure which describes where possible cavitation damage could occur by the collapse of bubbles in the pressure recovery areas of the flow control valves.

Full cavitation, however, is a very unlikely event in the FCV. In addition, even if full cavitation occurs in the FCV, it would take 4 to 8 hours before the 0.003 in. corrosion allowance is worn away locally.

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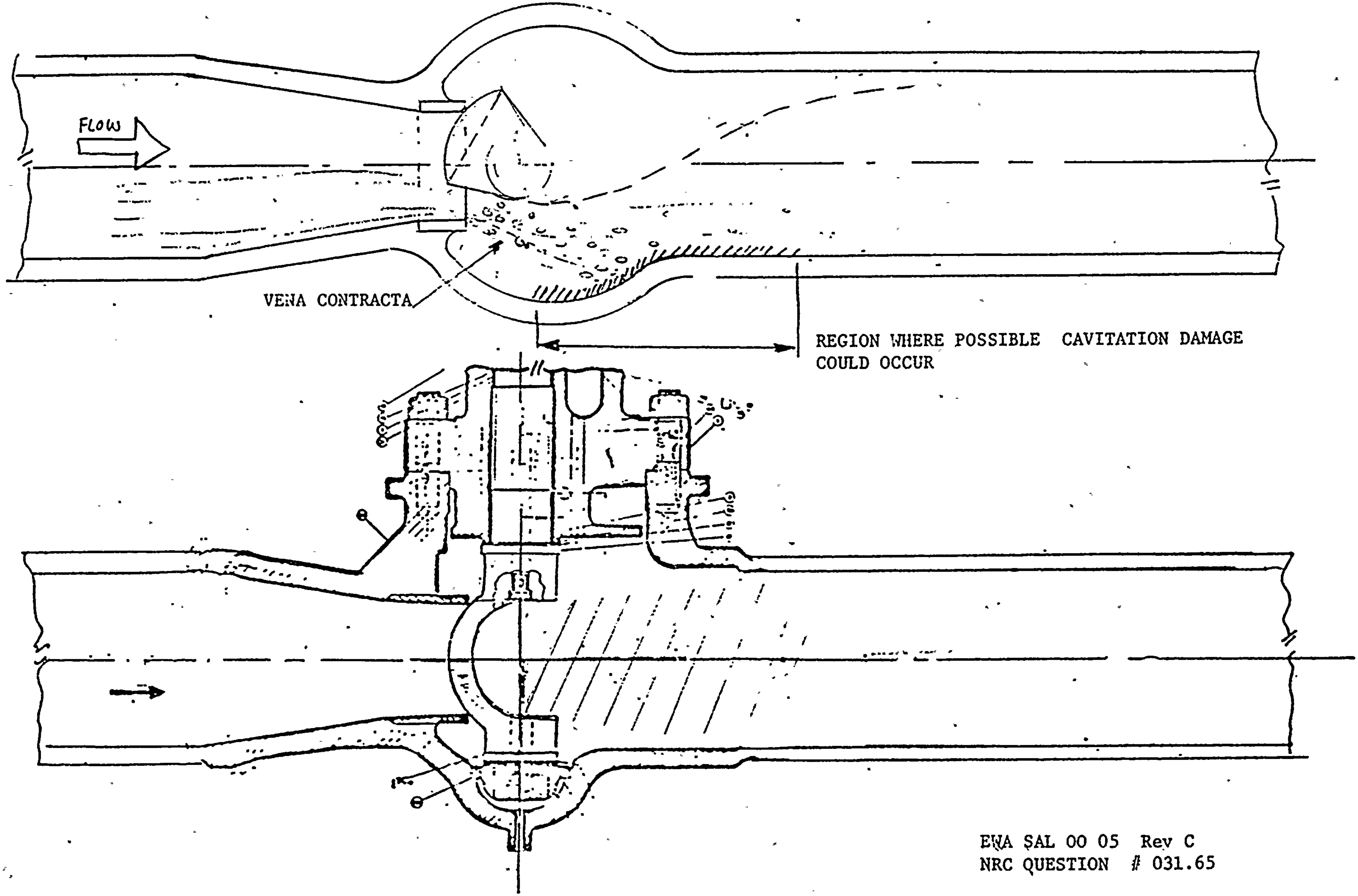

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NRC QUESTION # 031.65

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Q 31.066
(31.009)
(31.018)

Your response to Item 31.018 appears to be based on design features which are not part of the BWR-5 design. Accordingly, provide the following additional information to support the validity of your response:

a. Describe any differences between the control rod scram speed and the resultant reactivity insertion rate for the WNP-2 facility and those of the GESSAR-238 design (Docket No. STN 40-447).

b. Clarify any discrepancies between the relief valve system design described in your response to Item 31.009 and the Class 1E relief valve design which is proposed in the GESSAR-238 design.

c. Describe the effects of the differences in the core size and physical characteristics of the WNP-2 reactor core and those of the Zimmer facility, on the validity of referencing the Zimmer study.

Response:

The response to Item 31.018 included the BWR-6 GESSAR study to give an overall view of the results for different product lines. The NRC has concluded that the offsite doses calculated in the BWR-6 GESSAR study are well within 10CFR100 limits. However, a more valid comparison would be one to the Zimmer plant, another BWR-5.

Parts (a) and (b) of Question 31.066 query the differences in reactivity insertion rate and relief valve system design. WNP-2 has the same control rod drive speed and resultant reactivity insertion rate as the Zimmer plant. The relief valve system design for WNP-2 is also similar to Zimmer, with the safety-relief valve capacities being equivalent.

By analyzing the limiting pressurization transient with concurrent failures of direct scram, the RPT and the bypass functions, it has been demonstrated in response to NRC Question 221.359 on Zimmer Docket that the BWR 4/5/6 designs are each able to remain within the limits of 10CFR100. Even with widely varying design parameters, the results for each design were similar.

Thus, a like analysis solely for WNP-2 would provide results similar to that reached in the Zimmer and GESSAR study, i.e., that the offsite doses are well within 10CFR100 limits.

Q 31.067
(RSP)
(31.025)

Your response to Item 31.025 indicates that the single failure of instrument bus 1A will result in a total loss of direct indication to the operator of a safe reactor shutdown. It is the staff's position that this is a unacceptable design. Accordingly, we require you to provide a design which will satisfy the single failure criterion.

Response:

The system has been changed to provide essential power to the full core display and the scram valve position indicators. A power panel which is fed from a diesel generator upon loss of offsite power is now the source rather than instrument bus 1A. In addition, the operator has alternate diverse means of determining a reactor trip as follows:

1. Rod position display from the rod sequence control system.
2. Process computer trip and position log.
3. R.P.S. system annunciation.
4. Local mechanical pressure indicators for the scram valve pilot air header.
5. Neutron monitoring flux indicators.

The redesign and diverse means of verifying reactor shutdown meet the single failure criteria.

Q 31.068

Provide the wiring diagrams for the instrument racks for the WNP-2 facility which are comparable to the diagrams cited in the Item 031.032. It is the staff's position that these must be made available so that they may be audited to determine whether the required electrical separation of independent divisions has been achieved.

Response:

The following drawings are provided as requested in 31.068, (7 copies).*

Rack	Number	Sheet	Revision
IR-10	E538	7	3
IR-10	757-E-673		B
IR-11	E538	8	1
IR-11	757-E-674		E
H22-P004	E539	21	1
H22-P004	127D1827TC		2
H22-P005	E539	2	2
H22-P005	127D1837TC		2
H22-P026	E539	4	2
H22-P026	828E390TC		2
H22-P027	E539	3	1
H22-P027	828E387TC		2

* Packaged and sent separately.

Q. 031.069
(3.8.2.1)
(6.2.1.1)
(031.001)

Your response to Item 031.001(p) is incomplete. Describe the air supply, pressure control, and position indication for the butterfly valves in accordance with the guidance provided in Section 7.3.1 of Regulatory Guide 1.70. Clarify the reference to 6.2.1.1.2 in the response to Item 031.001(p) since this response does not address the staff's concern regarding the position indication instrumentation.

Response:

Please refer to 3.8.2.1.3, 6.2.1.1.2.c and 7.3.1.1.2.9.1 for the information requested.

Amendment 10, revising Chapter 7, deleted 7.3.1.1.2.9.1 as this section only referenced 3.8.2.1.3 and provided no other information.

Q 31.070
(RSP)
(6.2.2.2)
(6.5.2.2)
(7.3.1.1)
(031.001)
(031.011)

It is the staff's position that insufficient time is available for the operator to reliably take the manual actions which are necessary to initiate suppression pool spray during a small break. The staff has established the requirement for automatic initiation of suppression pool spray for the Mark II containment. Accordingly, we require you to provide a Class IE automatic control system for each suppression pool spray system.

Response

The WNP-2 design meets the intent of the proposed CSB Branch Technical Position on "Steam Bypass for Mark II Containments".

The history of the questions of steam bypass on WNP-2 are extensive, dating back to January 1972. Questions 5.4, 5.22, and 5.24 to the PSAR all respond to the concern. The SER (pp 63-65) summarized the NRC position on the issue at the CP stage and noted that WPPSS agreed to study additional means to mitigate the consequences or minimize the potential for bypass leakage. This was formally documented as a Post CP item in the notes of a NRC-WPPSS meeting held on October 17-18, 1973 (Reference 1). In the notes WPPSS committed to submitting a report on the matter. In August 1974, Reference 2 transmitted the WPPSS report WPPSS-74-2-R5, "Drywell to Wetwell Leakage Study", satisfying the commitment. The NRC requested additional information concerning the report in Reference (3). References (4) and (5) provide WPPSS responses to the NRC questions. Reference (6) indicated that Structural Engineering Branch found the applicable WPPSS responses acceptable. WPPSS has no record of feedback from Containment Systems Branch on the responses to its questions but assumed in Reference (5) that, in the absence of feedback, the post CP item was resolved. Accordingly, WPPSS has gone ahead with construction in these areas based on the above correspondence.