50-397

Results of

Hardness Testing and Metallurgical Examination

on

Feedwater Pipe Welds

at

Washington Nuclear Project No. 2 (WNP-2)

Report No. IE-119, September 28, 1979

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REPORT NO. 78206

CLIENT: PARAMETER, INC.

SUBJECT: NRC - TASK - 06
CONTRACT NO. 05-77-186

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Date: September 28, 1979

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I. DESCRIPTION AND PURPOSE:

NRC Task Order 06 was initiated to determine whether or not weld No. 7 on isometric drawing RFW-418-4 and No. 7 on isometric drawing RFW-419-4 were subject to radically excessive and non-uniform post weld heat treatment. Both welds are entitled "Reactor Feedwater from Flowmeter to Reactor Vessel" and are part of Washington Nuclear Project No. 2 (WNP-2) in Benton County, Washington. Technimet Corporation performed in situ hardness tests and microscopic examinations on fulfillment of this task with the results reported herein.

II. CONCLUSIONS:

A. No evidence of radically excessive or non-uniform post weld heat treatment was found. The maximum temperature reached at any point estimated to be in the range of 1270-1310° F.

Some non-uniformity was noted but the maximum temperature differential around the perimeter of the weld zone and pipe is estimated to be less than 100°F.

B. A weld repair was noted on the valve body of Valve Train A. This area has a well defined heat affected zone evidencing that the repair was made after the quench and temper treatment of the valve and further indication that post weld heat treatment did not exceed the lower critical temperature, e.g. approximately 1330° F.

III. PRELIMINARY TESTS:

Prior to performing on-site inspection, preliminary tests including calibration and equipment evaluation were performed.

The instruments used in preliminary and on-site testing are listed in Appendix A along with their calibration. In addition to the listed equipment, a Wilson Model M-9 portable hardness tester was evaluated. This tester was not used by mutual agreement with Mr. Joe Collins - NRC for the reasons outlined in Technimet's September 10, 1979 preliminary report to Parameter, Inc. (Attached to Appendix A).

The Equotip hardness tester was evaluated using 3 test blocks developed by Technimet. The hardnesses and description of these blocks is contained in Appendix B. Our tests indicate the Equotip tester provides accuracy within scale conversion variations in the range of DPH 144-286. This, along with ease of operation and rapidity of testing made this unit the logical choice.

Full scale tests on a 24" diameter ASME-SA106-GRB tube were performed to further prove the test procedures. The results recorded in Appendix C indicated good consistency and equipment capability.

IV. ON-SITE TESTS:

On-site tests were performed September 12-14, 1979, with periodic observation by NRC Mssrs. Tom Bishop and Al Toth. The test locations are keyed to Fig. 1 and the test procedures are illustrated in Figs. 2 and 3. According to the sketch provided by Mssr. Bishop, the strip heaters were adjacent to but not extending over the weld area and the controlling thermocouple were at the center of the welds. Consequently, we would anticipate any overheating to be most severe near positions VB 2 and P2 because these positions are located directly under the heaters and a thermal gradient undoubtedly existed.

IV. ON-SITE TESTS: (Continued)

A. Hardness Tests:

The hardness test results are summarized in Tables 1 and 2 with Tables 3-17 providing the recorded data. In comparing the hardness of the valve body in an area away from weld to the post heat treated region it is apparent that some softening occurred during post weld heat treating. The most dramatic drop in hardness (about 40 points BHN) occurred in valve train B near the 180° position. Valve train A reflects a less severe hardness reduction also occurring near the 180° position. Since the valve is a quenched and tempered structure reportedly furnace tempered 4.0 hours at 1220° F., it is concluded that the post weld treatment exceeded this temperature. Based on a normal tempering response and a 3.0 hour treatment, we estimate that the post weld heat treatment temperature was in the range of 1270-1310° F. A similar trend is evident on the pipe side of the weld where a softening of up to 30 points BHN appears to have occurred.

Examination of the hardness results around the periphery indicates some non-uniformity of heating. This is not unusual or unexpected for the heating method reportedly used. The degree of non-uniformity is not severe and we find no indication of extreme temperature gradients in excess of 50-100° F.

B. Microscopic Examination:

Photomicrographs of suitably prepared surfaces etched with 2% nital were taken at various distances from the weld at 0° and 180°. These are shown in Figures 4-20 and reflect no evidence of gross overheating. All the areas reviewed reflected normal, fine grained structures.

IV. ON-SITE TESTS: (Continued)

B. Microscopic Examination: (Continued)

The pipe itself indicated a hot rolled and normalized steel while the valve body reflects a quenched and tempered structure. Grain size of all areas was ASTM E112 6-8 (fine) except immediately adjacent to welds where size 4 existed as a natural consequence of welding. Post heat treatment had no effect on grain size.

We located a repair weld in Valve A at the VB2 (60°) position with its fusion line approximately parallel to the chamfer and located 8¼" from the weld centerline. The structure of the repaired area is shown in Figs. 5-7. If this area had been heated to above the lower critical temperature (approx. 1330° F.) the fusion line of this weld would have started to form new grains and become obliterated. It is concluded that the weld was made after the valve was quenched and tempered and that the area never exceeded 1330° F. after the weld repair was in place.

V. SUMMARIZING:

In summary, our tests indicate no evidence of radically excessive post weld heat treatment. Although post weld heat treatment temperature may have reached 1310° F. in some local areas, no metallurgical damage is apparent and the tensile properties in all areas of the valve are expected to meet SA-216 Gr WCB. Similarly, hardness and microstructure of the pipe is such that we expect it to meet the tensile requirements of SA-106.

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Respectfully submitted,

TECHNIMET CORPORATION

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MES/mp
Attachments

TABLE 1

HARDNESS TEST RESULTS - "L" VALUES

Ref: Drawing No. RFW-418-4 Weld No. 7

LOCATION PER FIGURE 1

POS	VB1	VB2	VB3	VB4	W (Weld)	P4	P3	P2	P1
See Sketch	480.5 BHN=208								444.9 BHN=178
0°			466.1 BHN=195	449.3 BHN=182	431.8 BHN=167	407.3 BHN=149	415.3 BHN=154	424.7 BHN=162	
60°		Valve Body: 475.0 BHN=203; Repair Weld: 452.3 BHN=186	455.5 BHN=186	454.5 BHN=186	422.2 BHN=160	407.9 BHN=149	412.2 BHN=152	420.6 BHN=158	
120°			466.3 BHN=195	460.1 BHN=190	455.2 BHN=186	411.4 BHN=152	413.5 BHN=153	423.5 BHN=161	
180°			456.8 BHN=188	440.5 BHN=174	437.2 BHN=172	404.5 BHN=147	406.2 BHN=148	421.7 BHN=160	
240°			460.7 BHN=191	470.8 BHN=199	471.7 BHN=200	403.2 BHN=146	416.5 BHN=155	423.3 BHN=161	
300°		438.3 BHN=173	463.7 BHN=194	456.8 BHN=187	440.3 BHN=174	406.1 BHN=148	409.2 BHN=151	423.2 HN=161	

NOTE: Hardness in Equotip "L" values except when converted to BHN as noted.

TABLE 2

HARDNESS RESULTS - SUMMARY

Valve Train "B"
Ref: Drawing No. RFW - 419-4 Weld No. 7
LOCATION PER FIGURE 1

POS	VB 1	VB2	VB3	VB4	W (Weld)	P4	P3	P2	P1
See Sketch	468.3 BHN=197	Not Tested							Not Tested
0°			427.9 BHN=164	426.5 BHN=163	406.8 BHN=148	408.7 BHN=150	416.4 BHN=155	415.2 BHN=154	
60°			456.2 BHN=187	445.2 BHN=178	412.2 BHN=152	411.0 BHN=151	415.0 BHN=154	425.9 BHN=162	
120°			439.9 BHN=174	439.7 BHN=174	420.8 BHN=158	415.6 BHN=154	413.9 BHN=153	420.4 BHN=158	
180°			419.0 BHN=158	415.7 BHN=154	402.9 BHN=145	407.5 BHN=149	402.0 BHN=145	419.7 BHN=158	
240°			419.2 BHN=158	418.3 BHN=156	407.6 BHN=148	402.8 BHN=145	411.5 BHN=152	416.9 BHN=156	
300°			430.7 BHN=166	429.0 BHN=166	417.2 BHN=156	414.5 BHN=154	414.5 BHN=154	419.4 BHN=158	

NOTE: Hardnesses in Equotip "L" values except when converted to BHN as noted.

TABLE 3

HARDNESS TEST RESULTS

Valve Train "A"
Ref: Dwg. No. RFW-418-4 Weld No. 7

LOCATION PER FIG. 1

Reading	VB 1	VB 2 @60°(Body)	VB 2 @60°(Repair Weld)	VB 2 @300°
1	489	504	467	428
2	494	477	464	436
3	491	493	464	432
4	493	495	459	452
5	496	469	450	450
6	493	466	438	453
7	494	468	473	444
8	464	464	459	455
9	489	463	462	426
10	495	471	465	452
11	492	481	458	439
12	495	480	462	452
13	491	489	465	453
14	489	505	468	459
15	493	500	431	442
TOTAL	7358	7225	6885	6675
AVERAGE	490.5	481.7	459.0	445.0
POSITION CORRECTION	-10.0	-6.7	-6.7	-6.7
CORRECTED AVERAGE	480.5	475.0	452.3	438.3
EQUIVALENT BHN	208	203	186	173

TABLE 4

HARDNESS TEST RESULTS

Valve Train A
Ref: Dwg. No. RFW-418-4 Weld No. 7

LOCATION PER FIG. 1

Reading	VB3@ 0°	VB3@60°	VB3@120°	VB3@180°	VB3@240°	VB3@300°
1	463	464	478	472	482	463
2	480	460	476	473	479	462
3	463	462	477	485	468	470
4	470	461	481	484	472	471
5	462	467	477	481	478	476
6	469	462	480	481	472	475
7	466	463	478	476	477	470
8	470	462	503	471	470	479
9	464	459	489	471	476	472
10	462	462	476	481	469	465
11	463	465	480	477	486	465
12	463	460	476	484	479	470
13	462	463	475	477	480	475
14	469	463	474	478	464	468
15	465	460	485	476	468	475
TOTAL	6991	6933	7205	7167	7120	7056
AVERAGE	466.1	462.2	480.3	477.8	474.7	470.4
POSITION CORRECTION	0	-6.7	-14.0	-21.0	-14.0	-6.7
CORRECTED AVERAGE	466.1	455.5	466.3	456.8	460.7	463.7
EQUIVALENT BHN	195	186	195	188	191	194

TABLE 5

HARDNESS TEST RESULTS

Valve Train A
Ref: Dwg. No. RFW-418-4 Weld No. 7

LOCATION PER FIG. 1

Reading	VB4@0°	VB4@60°	VB4@120°	VB4@180°	VB4@240°	VB4@300°
1	433	459	472	432	482	451
2	458	466	478	426	473	455
3	450	457	475	446	480	452
4	456	458	474	459	477	454
5	453	461	474	479	480	461
6	453	460	472	439	471	466
7	448	468	473	459	509	466
8	435	460	472	481	473	463
9	451	461	462	482	479	466
10	466	463	473	475	473	477
11	456	459	478	474	493	476
12	451	456	478	471	470	468
13	447	463	477	478	509	463
14	452	467	474	453	476	468
15	451	460	480	471	527	466
SUM.	6740	6918	7112	6923	7272	6952
AVERAGE	449.3	461.2	474.1	461.5	484.8	463.5
POSITION CORRECTION	0	-6.7	-14.0	-21.0	-14.0	-6.7
CORRECTED AVERAGE	449.3	454.5	460.1	440.5	470.8	456.8
EQUIVALENT BHN	182	186	190	174	199	187

TABLE 6

HARDNESS TEST RESULTS

Valve Train A
Ref: Dwg. No. RFW-418-4 Weld No: 7

LOCATION PER FIG. 1

Reading	W@0°	W@60°	W@120°	W@180°	W@240°	W@300°
1	446	425	443	470	478	448
2	418	441	464	484	507	446
3	409	426	473	438	468	443
4	418	432	460	439	508	461
5	443	433	473	466	492	441
6	457	433	482	470	471	450
7	410	432	459	448	491	433
8	418	422	465	442	486	458
9	420	436	476	458	482	447
10	445	426	473	470	515	452
11	447	436	477	468	486	442
12	431	420	472	457	484	443
13	436	428	478	433	463	444
14	433	428	475	466	482	452
15	446	415	470	464	483	445
SUM	6477	6433	7038	6873	7286	6705
AVERAGE	431.8	428.9	469.2	458.2	485.7	447.0
POSITION CORRECTION	0	-6.7	-14.0	-21.0	-14.0	-6.7
CORRECTED AVERAGE	431.8	422.2	455.2	437.2	471.7	440.3
EQUIVALENT BHN	167	160	186	172	200	174

TABLE 7

HARDNESS TEST RESULTS

Valve Train A

Ref: Dwg. No. RFW-418-4 Weld No. 7

LOCATION PER FIG. 1

Reading	P4@0°	P4@60°	P4@120°	P4@180°	P4@240°	P4@300°
1	416	421	426	429	426	411
2	409	414	426	427	420	413
3	404	417	425	423	415	402
4	406	412	447	427	421	406
5	402	414	423	421	419	409
6	404	412	415	427	414	420
7	407	414	415	423	411	412
8	403	411	427	423	419	424
9	404	411	425	429	421	418
10	406	413	426	431	422	405
11	416	419	427	428	415	417
12	414	415	435	429	416	408
13	408	416	426	424	419	405
14	409	414	423	431	417	415
15	402	416	425	428	413	427
SUM	6110	6219	6391	6398	6268	6192
AVERAGE	407.3	414.6	426.1	426.5	417.9	412.8
POSITION CORRECTION	0	-6.7	-14.7	-22.0	-14.7	-6.7
CORRECTED AVERAGE	407.3	407.9	411.4	404.5	403.2	406.1
EQUIVALENT BHN	149	149	152	147	146	148

TABLE 8

HARDNESS TEST RESULTS

Valve Train A

Ref. Dwg. No. RFW-418-4 Weld No. 7

LOCATION PER FIG. 1

Reading	P3@0°	P3@60°	P3@120°	P3@180°	P3@240°	P3@300°
1	420	417	418	401	428	426
2	402	436	428	426	442	415
3	432	413	421	423	427	412
4	423	421	422	446	426	417
5	421	422	429	422	428	421
6	416	412	432	425	428	420
7	404	417	417	431	421	412
8	412	409	434	428	433	404
9	416	425	443	436	427	416
10	413	425	420	432	438	415
11	419	423	432	425	437	418
12	411	416	460	438	438	416
13	412	416	426	429	441	415
14	409	418	425	424	433	415
15	420	414	416	437	420	417
SUM	6230	6284	6423	6423	6468	6239
AVERAGE	415.3	418.9	428.2	428.2	431.2	415.9
POSITION CORRECTION	0	-6.7	-14.7	-22.0	-14.7	-6.7
CORRECTED AVERAGE	415.3	412.2	413.5	406.2	416.5	409.2
EQUIVALENT BHN	154	152	153	148	155	151

TABLE 9

HARDNESS TEST RESULTS

Valve Train A

Ref: Dwg. No. RFW-418-4 Weld No. 7

LOCATION PER Fig. 1

Reading	P2@0°	P2@60°	P2@120°	P2@180°	P2@240°	P2@300°
1	425	419	444	459	441	446
2	416	433	440	436	433	428
3	441	443	440	453	448	423
4	418	423	437	459	434	423
5	435	422	436	441	445	437
6	421	420	432	446	460	436
7	417	418	447	444	432	417
8	414	414	430	434	435	429
9	428	418	438	443	434	435
10	420	434	438	444	433	440
11	419	435	435	423	431	435
12	417	442	454	445	444	433
13	437	428	431	439	428	421
14	430	421	429	452	436	423
15	432	439	442	437	436	421
SUM	6370	6409	6573	6655	6570	6449
AVERAGE	424.7	427.3	438.2	443.7	438.0	429.9
POSITION CORRECTION	0	-6.7	-14.7	-22.0	-14.7	-6.7
CORRECTED AVERAGE	424.7	420.6	423.5	421.7	423.3	423.2
EQUIVALENT BHN	162	158	161	160	161	161

TABLE 10

HARDNESS TEST RESULTS

Valve Train A

Ref: Dwg. No. RFW-418-4 Weld No. 7

LOCATION PER FIG. 1

Reading	Vertical Riser 7-8" above elbow weld
1	464
2	460
3	454
4	460
5	458
6	455
7	459
8	463
9	478
10	451
11	439
12	438
13	442
14	448
15	454
SUM	6823
AVERAGE	454.9
POSITION CORRECTION	-10.0
CORRECTED AVERAGE	444.9
EQUIVALENT BHN	178

TABLE 11

HARDNESS TEST RESULTS

Valve Train B

Ref: Dwg. No. RFW-419-4 Weld No. 7

LOCATION PER FIG. 1

Reading	VB1 (Valve Body at Top Flange)
1	472
2	480
3	482
4	478
5	474
6	479
7	478
8	477
9	479
10	480
11	473
12	482
13	479
14	480
15	482
SUM	7175
AVERAGE	478.3
POSITION CORRECTION	-10.0
CORRECTED AVERAGE	468.3
BHN EQUIVALENT	197

TABLE 12

HARDNESS TEST RESULTS

Valve Train B
Ref: Dwg. No. RFW-419-4 Weld No. 7

LOCATION PER FIG. 1

Reading	VB3@0°	VB3@60°	VB3@120°	VB3@180°	VB3@240°	VB3@300°
1	429	456	452	441	433	439
2	424	479	443	447	434	433
3	425	465	452	444	431	436
4	427	464	449	438	439	433
5	420	457	440	444	432	433
6	424	456	440	439	435	434
7	429	448	486	440	435	438
8	429	469	460	438	427	437
9	438	457	450	438	431	440
10	434	460	450	439	433	447
11	429	480	456	441	435	436
12	428	466	449	438	430	437
13	430	460	475	437	433	439
14	427	446	459	437	436	440
15	426	474	448	440	434	439
SUM	6419	6943	6809	6600	6498	6561
AVERAGE	427.9	462.9	453.9	440.0	433.2	437.4
POSITION CORRECTION	0	-6.7	-14.0	-21.0	-14.0	-6.7
CORRECTED AVERAGE	427.9	456.2	439.9	419.0	419.2	430.7
BHN EQUIVALENT	164	187	174	158	158	166

TABLE 13

HARDNESS TEST RESULTS

Valve Train B
Ref: Dwg. No. RFW-419-4 Weld No. 7

LOCATION PER FIG. 1

Reading	VB4@0°	VB4@60°	VB4@120°	VB4@180°	VB4@240°	VB4@300°
1	426	447	451	436	431	439
2	427	451	465	434	432	439
3	428	454	450	424	435	435
4	427	448	451	439	434	432
5	426	446	470	439	435	429
6	429	445	453	434	435	438
7	429	447	452	438	438	437
8	426	448	451	436	438	433
9	425	461	449	440	426	419
10	429	452	442	437	424	437
11	426	463	446	439	435	441
12	423	449	455	435	429	438
13	427	464	452	440	431	438
14	430	448	459	438	431	440
15	420	458	455	442	430	440
SUM	6398	6778	6801	6551	6484	6535
AVERAGE	426.5	451.9	453.4	436.7	432.3	435.7
POSITION CORRECTION	0	-6.7	-14.0	-21.0	-14.0	-6.7
CORRECTED AVERAGE	426.5	445.2	439.4	415.7	418.3	429.0
BHN EQUIVALENT	163	178	174	154	156	166

TABLE 14

HARDNESS TEST RESULT

Valve Train B
Ref: Dwg. No. RFW-419-4 Weld No. 7

LOCATION PER FIG. 1

Reading	W@0°	W@60°	W@120°	W@180°	W@240°	W@300°
1	409	412	430	430	426	419
2	412	416	437	429	412	420
3	401	419	432	419	424	420
4	413	418	427	418	421	421
5	401	440	440	416	410	425
6	416	435	444	423	430	426
7	415	414	434	424	426	422
8	404	420	439	431	427	424
9	417	421	424	437	413	436
10	408	430	429	428	418	427
11	396	422	439	424	421	428
12	408	416	440	402	430	423
13	399	419	448	448	428	417
14	395	414	440	421	422	424
15	408	413	436	423	427	426
SUM	6102	6283	6539	6373	6335	6358
AVERAGE	406.8	418.9	435.5	424.9	422.3	423.9
POSITION CORRECTION	0	-6.7	-14.7	-22.0	-14.7	-6.7
CORRECTED AVERAGE	406.8	412.2	420.8	402.9	407.6	417.2
BHN EQUIVALENT	148	152	158	145	148	156

TABLE 15

HARDNESS TEST RESULT

Valve Train B
Ref: Dwg. No. RFW-419-4 Weld No. 7

LOCATION PER FIG. 1

Reading	P4@0°	P4@60°	P4@120°	P4@180°	P4@240°	P4@300°
1	411	420	434	446	417	411
2	415	418	432	442	416	419
3	408	404	433	422	417	424
4	408	414	429	421	418	421
5	403	420	434	423	417	429
6	405	417	423	429	415	417
7	407	416	429	420	414	417
8	409	420	434	419	416	419
9	409	418	436	420	417	418
10	424	420	429	444	424	422
11	402	419	431	429	420	416
12	412	419	431	429	418	423
13	412	421	426	432	418	432
14	404	423	426	446	418	427
15	402	416	427	422	417	425
SUM	6131	6265	6454	6442	6262	6318
AVERAGE	408.7	417.7	430.3	429.5	417.5	421.2
POSITION CORRECTION	0	-6.7	-14.7	-22.0	-14.7	-6.7
CORRECTED AVERAGE	408.7	411.0	415.6	407.5	402.8	414.5
BHN EQUIVALENT	150	151	154	149	145	154

TABLE 16

HARDNESS TEST RESULT

Valve Train B.
Ref: Dwg. No. RFW-419-4 Weld No. 7

LOCATION PER FIG. 1

Reading	P3@0°	P3@60°	P3@120°	P3@180°	P3@240°	P3@300°
1	416	433	425	426	428	411
2	413	419	421	423	427	419
3	417	414	428	423	438	424
4	420	418	427	430	438	421
5	408	419	431	425	422	429
6	410	419	427	428	417	417
7	427	420	429	415	420	417
8	417	416	435	428	421	419
9	421	418	433	425	419	418
10	415	427	433	423	418	422
11	412	425	432	420	425	416
12	419	424	427	422	428	423
13	418	418	422	423	425	432
14	419	428	441	423	434	427
15	414	427	418	426	433	425
SUM	6246	6325	6429	6360	6393	6318
AVERAGE	416.4	421.7	428.6	424.0	426.2	421.2
POSITION CORRECTION	0	-6.7	-14.7	-22.0	-14.7	6.7
CORRECTED AVERAGE	416.4	415.0	413.9	402.0	411.5	414.5
BHN EQUIVALENT	155	154	153	145	152	154

TABLE 17

HARDNESS TEST RESULTS

Valve Train B
Ref: Dwg. No. RFW-419-4 Weld No. 7

LOCATION PER FIG. 1

Reading	P2@0°	P2@60°	P2@120°	P2@180°	P2@240°	P2@300°
1	410	443	432	443	425	418
2	416	439	445	444	435	439
3	403	431	430	447	427	418
4	413	424	435	450	430	426
5	406	430	430	429	441	430
6	412	437	426	436	429	427
7	408	419	443	435	438	447
8	411	423	429	449	416	427
9	412	432	430	445	439	422
10	430	450	429	444	431	422
11	423	437	443	437	434	430
12	416	427	437	441	445	417
13	432	424	435	443	423	427
14	416	441	448	444	429	414
15	420	432	435	438	432	428
SUM	6228	6489	6527	6625	6474	6392
AVERAGE	415.2	432.6	435.1	441.7	431.6	426.1
POSITION CORRECTION	0	-6.7	-14.7	-22.0	-14.7	-6.7
CORRECTED AVERAGE	415.2	425.9	420.4	419.7	416.9	419.4
BHN EQUIVALENT	154	162	158	158	156	158

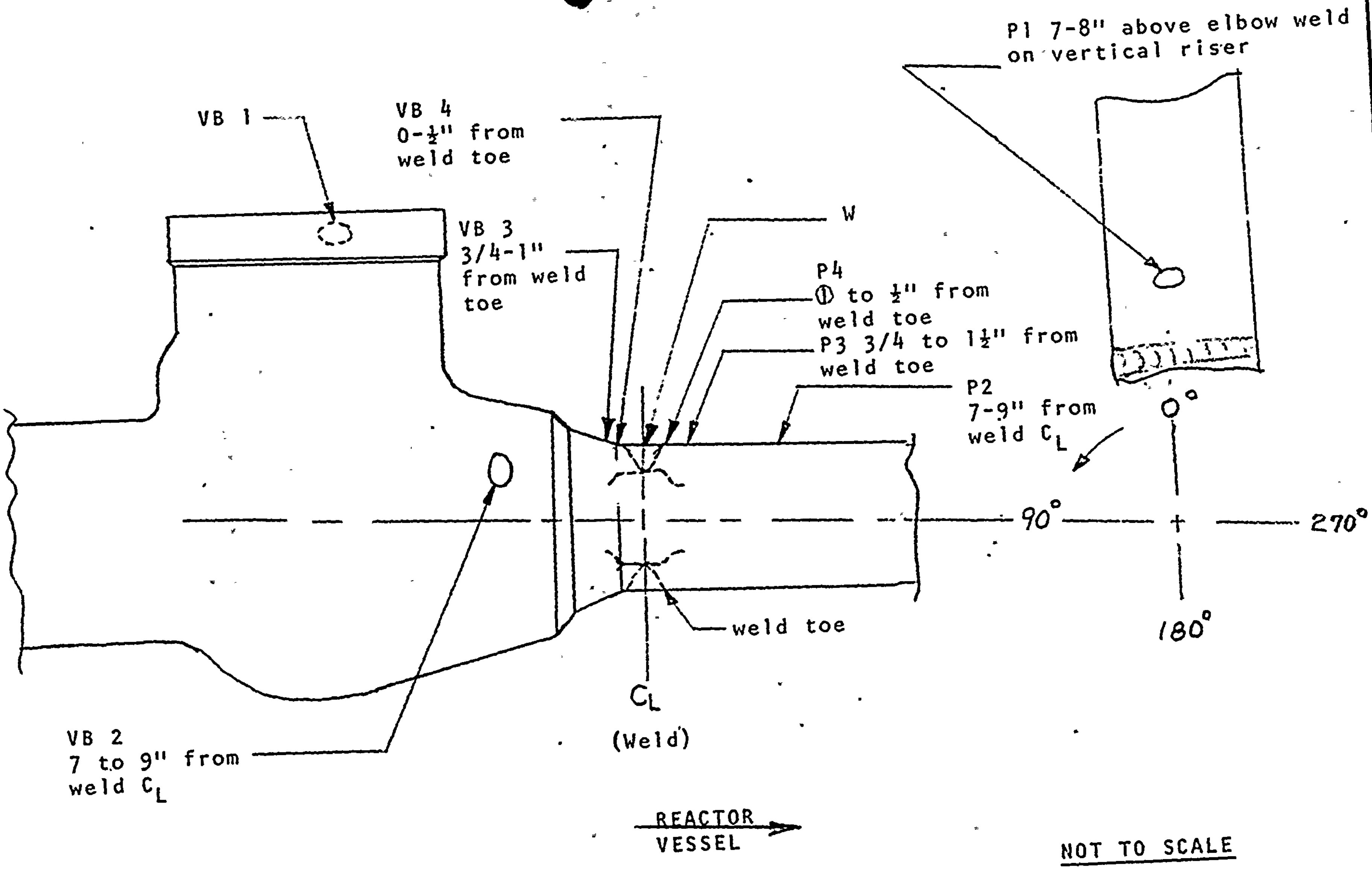

FIG. 1

Location of hardness tests and microstructure examinations.

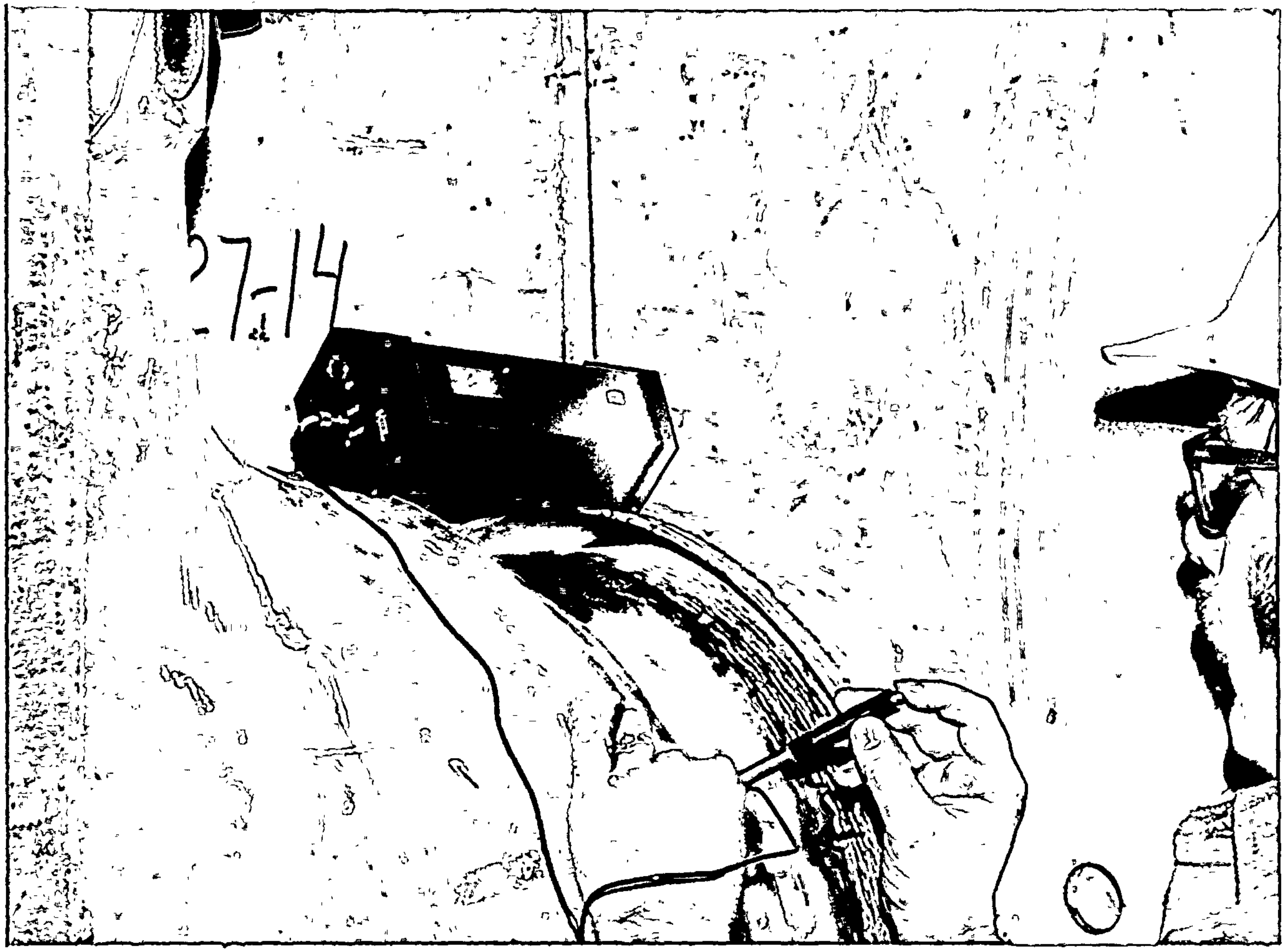

Fig. 2 - Hardness Testing at site.

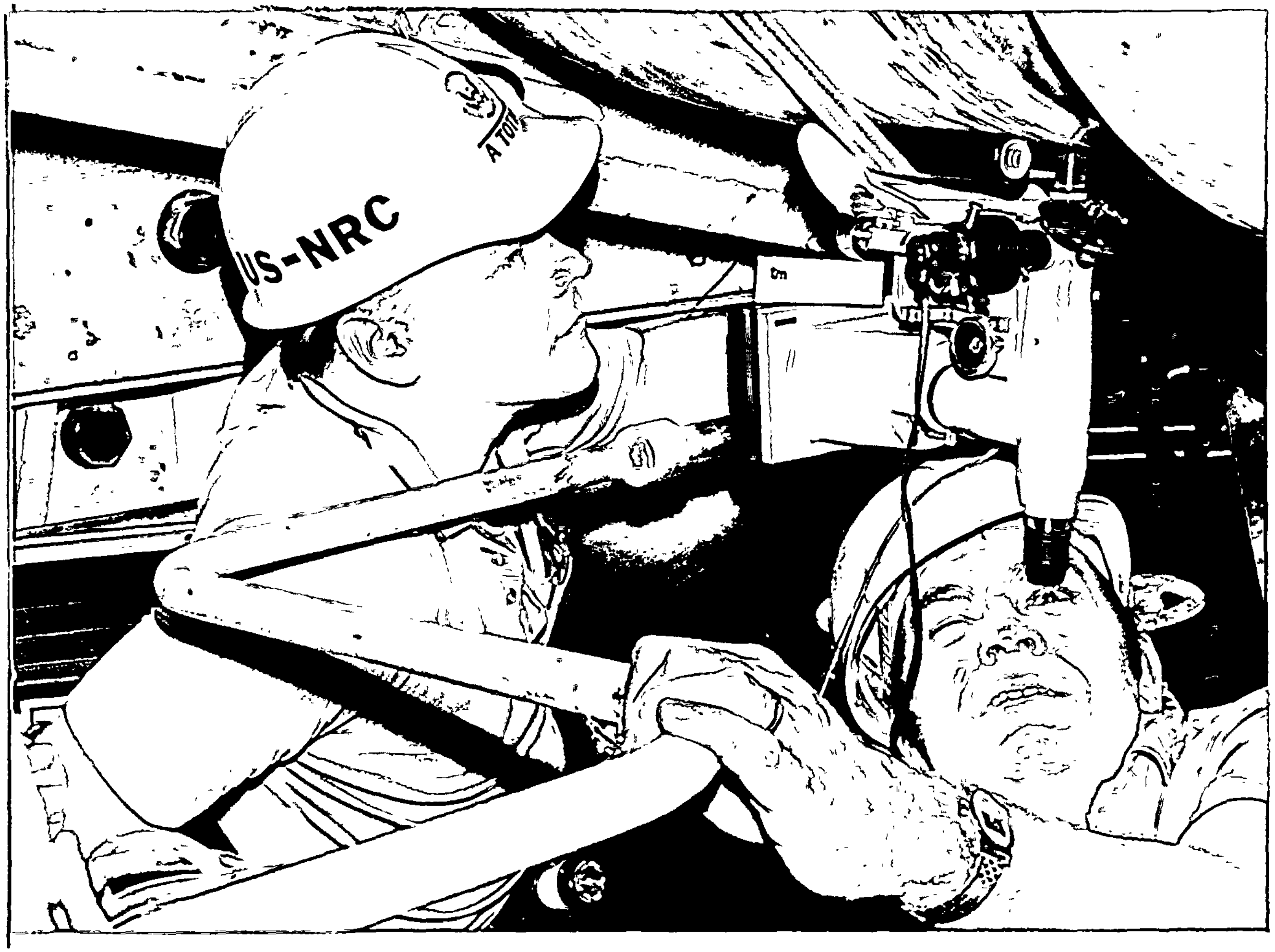

Fig. 3 - Microscopic examination at site.

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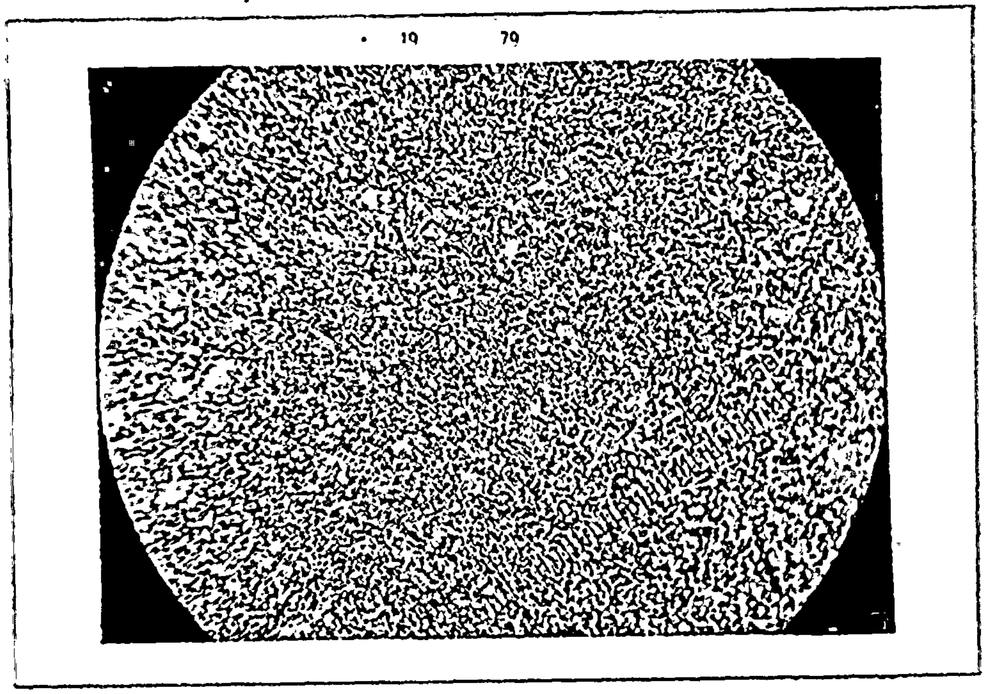

Fig. 4 - Line A - VB2 @ 60° - 8½" from weld #7 centerline. Structure is fine grained tempered martensite. 2% Nital. 100X

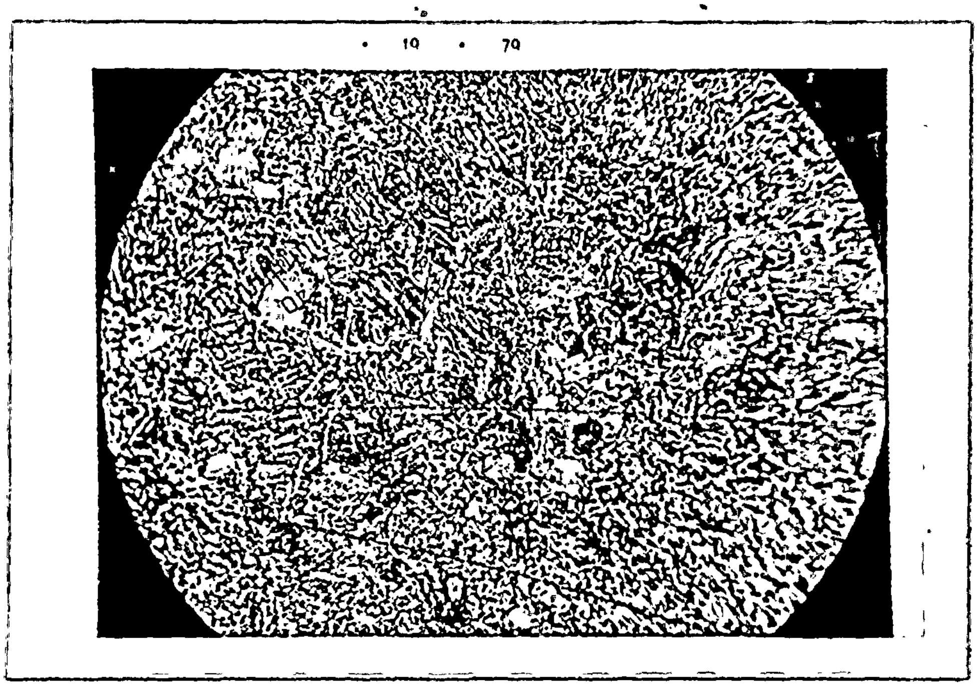

Fig. 5 - Line A - VB2 @ 60° - 8¼" from weld #7 centerline. Evidences a repair weld heat affected zone. Welding heat caused grain coarsening. Grain size ASTM E112 - 4 to 5. Structure indicates lower critical temperature was not exceeded during post weld heat treatment. 2% Nital. 100X.

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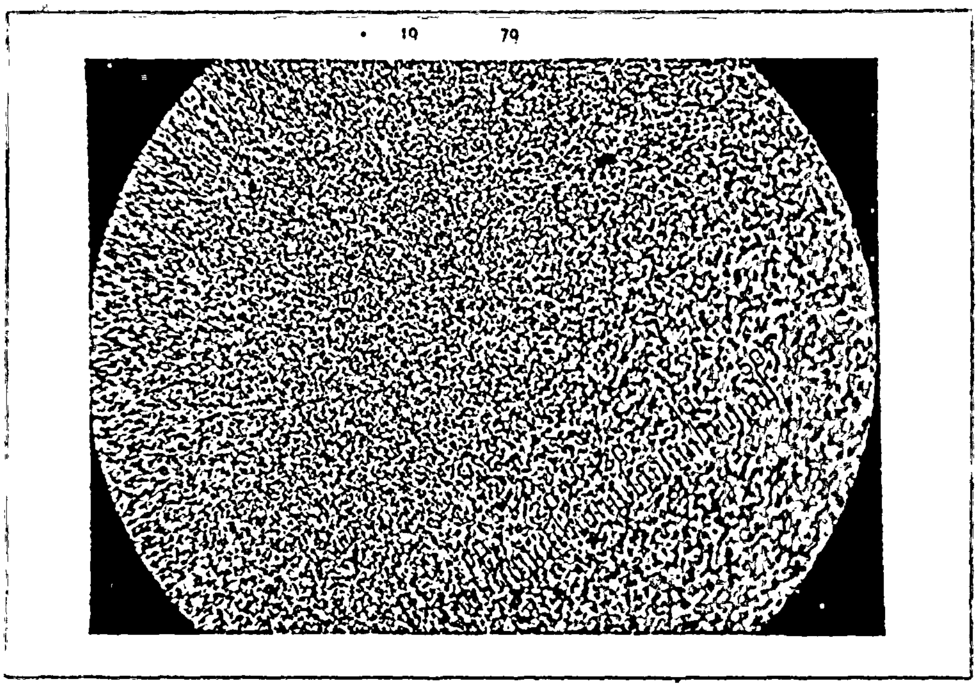

Fig. 6 - Area immediately adjacent to Fig. 5 where heat from welding resulted in grain refinement. 2% Nital. 100X.

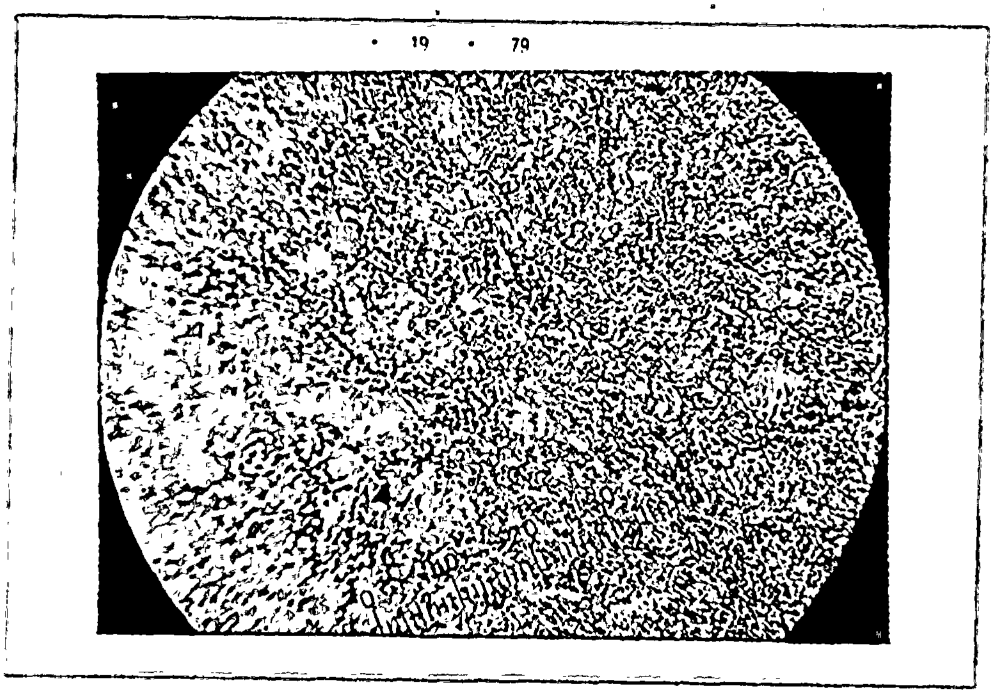

Fig. 7 - Area immediately adjacent to Fig. 6 reflecting typical weld metal structure consisting of bainite and ferrite. 2% Nital. 100X.

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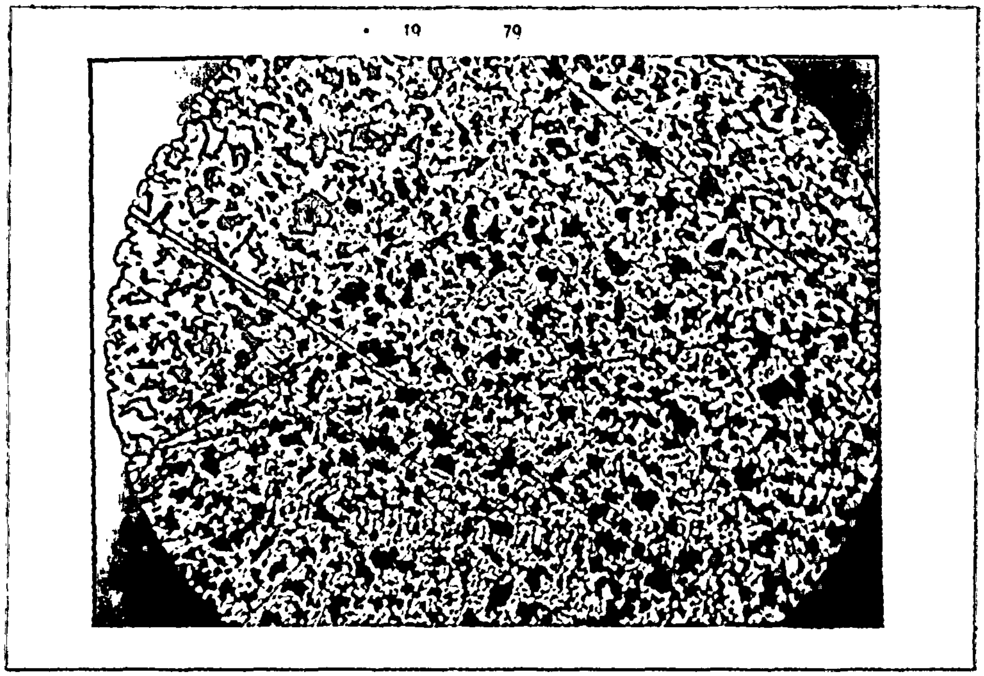

Fig. 8 - Line A - PI (Vertical Riser). Mixture of ferrite and pearlite typical of normalized or normalized and tempered condition. Grain size ASTME 112 7½-8. 2% Nital. 100X.

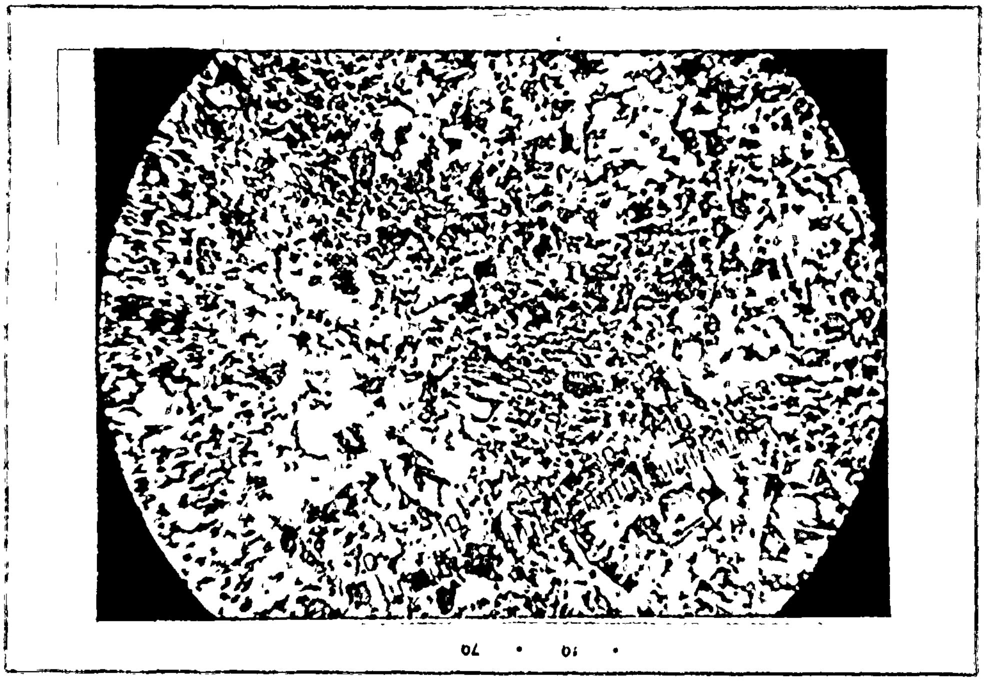

Fig. 9 - Line A - P2 at 0°. Mixture of pearlite and ferrite typical of normalized or normalized and tempered condition. Grain size ASTM E112 6-7. 2% Nital. 100X.

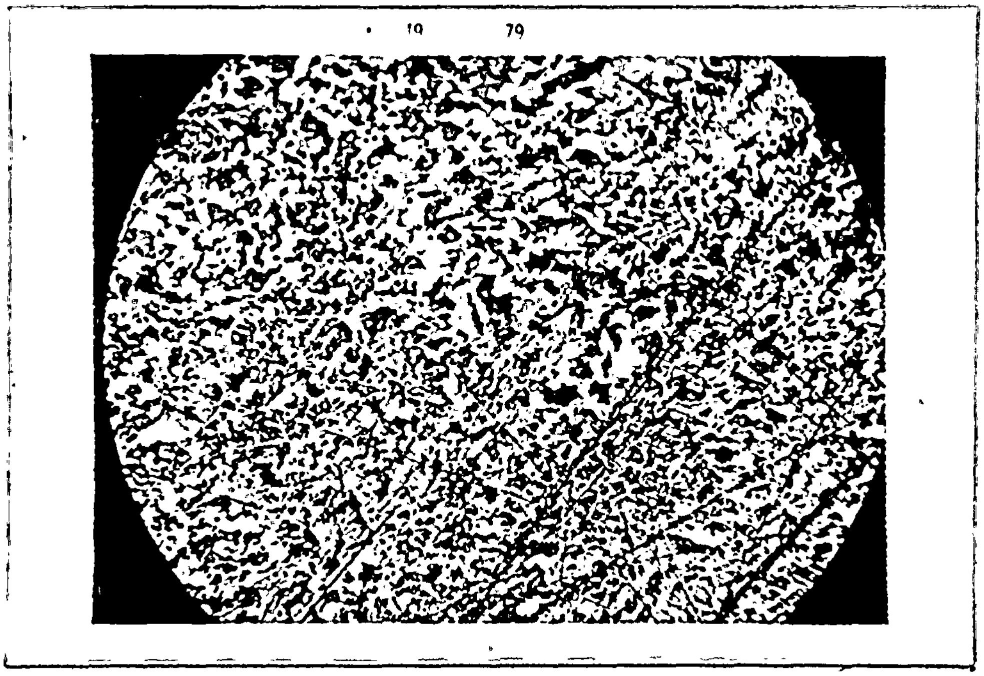

Fig. 10 - Line A - P2@ 180°. Structure is similar to Fig. 9 and reflects no radical difference in thermal history from 0°. Grain size ASTM E112 6-7. 2% Nital. 100X.

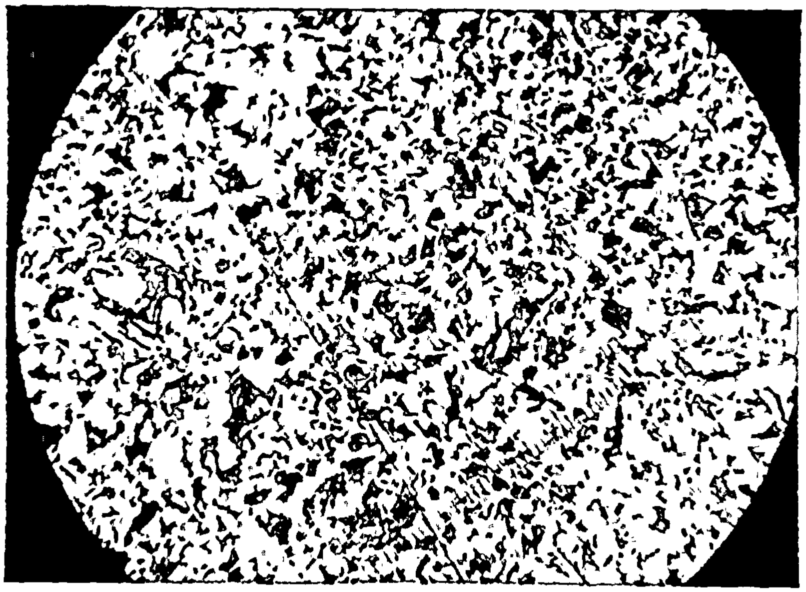

Fig. 11 - Line A - P4 @ 0^{o}. No change in structure from excessive heat is indicated. Grain size ASTM E112 7-7½. 2% Nital. 100X

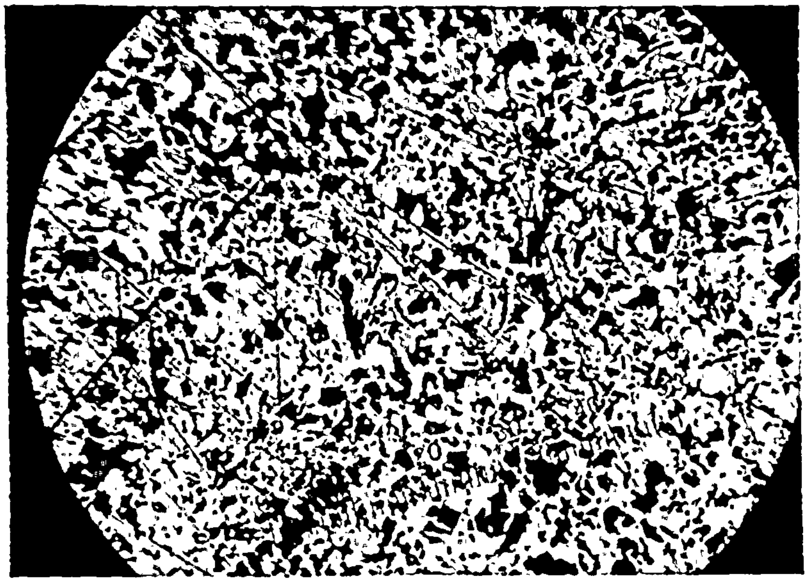

Fig. 12 - Line A - P4 @ 180^{o}. Essentially identical to Fig. 11 and normal. Grain Size ASTM E112 7-7½. 2% Nital. 100X.

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Fig. 13 - Line A - VB4 @ 0°. Reflects highly tempered, extremely fine grained martensite. Typical of expected structure in post weld heat treated zone. 2% Nital. 100X.

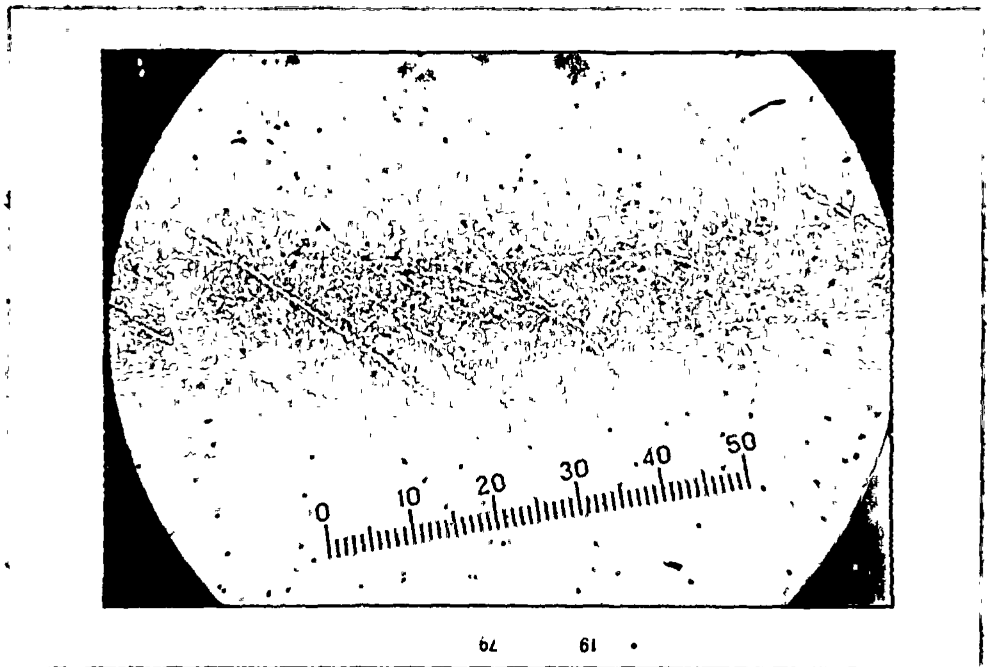

Fig. 14 - Line A - VB4 @ 180°. Essentially identical to Fig. 13. 2% Nital. 100X.

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Fig. 15 - Line B - P2 @ 0°. Fine grained ferrite and pearlite typical of normalized or normalized and tempered steel. Grain size ASTM E112 7-8. 2% Nital. 100X.

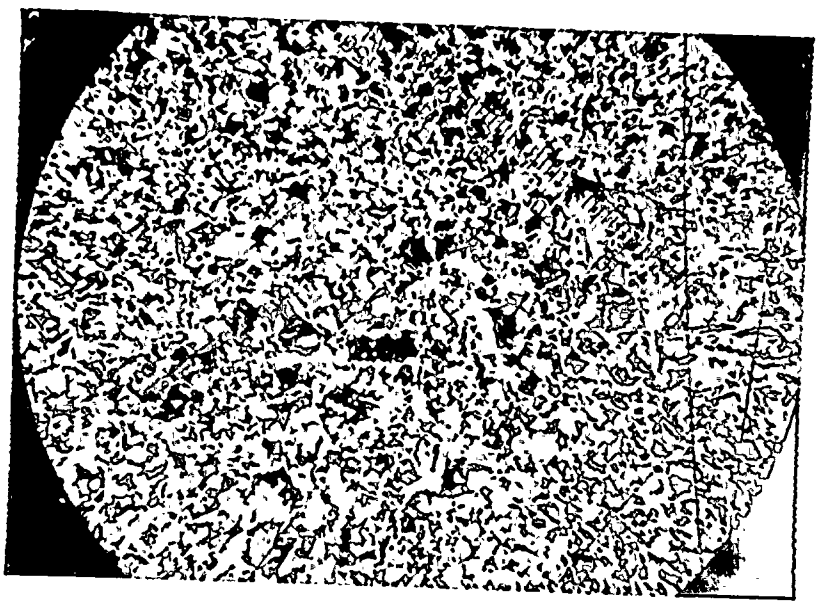

Fig. 16 - Line B - P2 @ 180°. Essentially identical in structure to Fig. 15. 2% Nital. 100X.

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Fig. 17 - Line B - P4 @ 0°. Fine grained ferrite and pearlite. Grain size ASTM E112 6-7. 2% Nital. 100X.

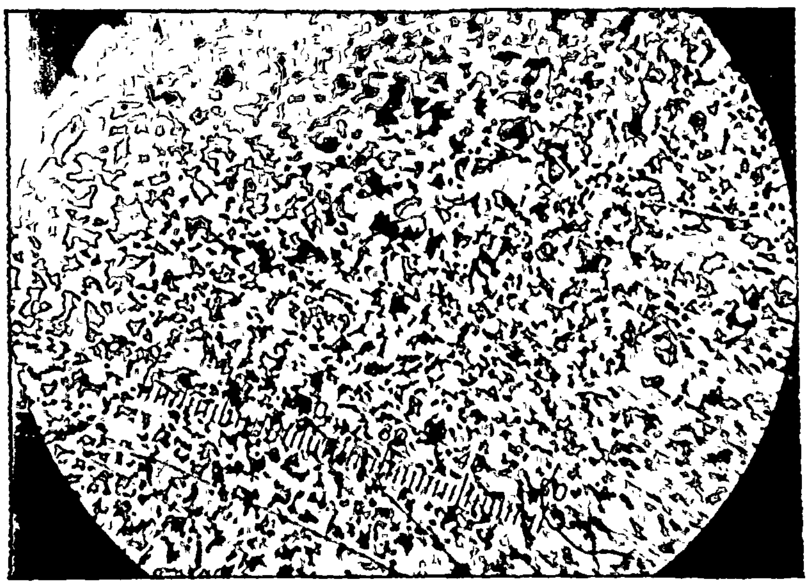


Fig. 18 - Line B - P4 @ 180°. Essentially identical to Fig. 17. 2% Nital. 100X.

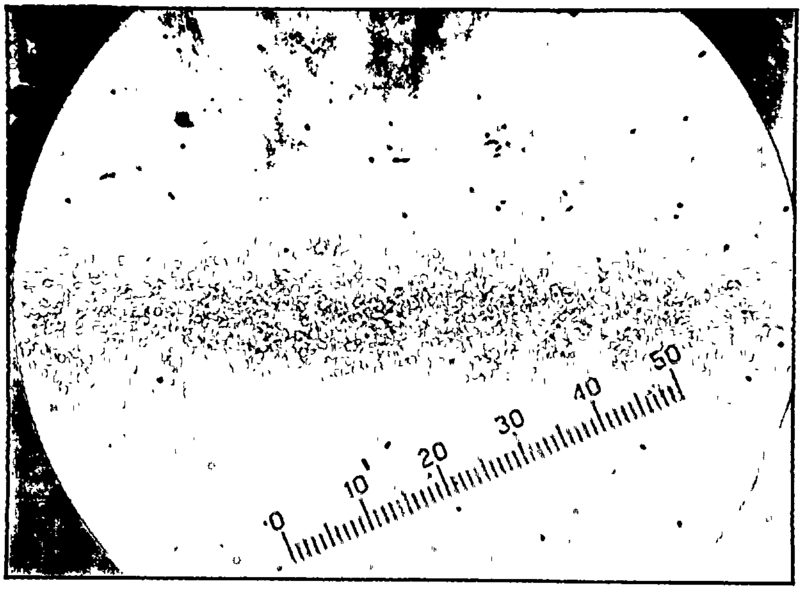


Fig. 19 - Line B - VB4 @ 0°. Fine grained, highly tempered martensite as expected. 2% Nital. 100X.

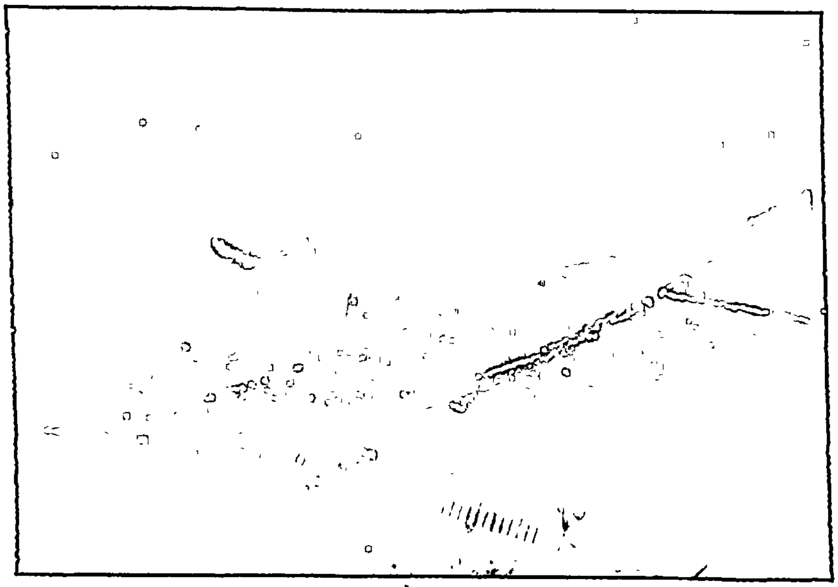


Fig. 20 - Line B - VB4 @ 180°. Essentially identical to Fig. 19 indicating no significant difference in thermal history. 2% Nital. 100X.

APPENDIX A

LIST OF INSTRUMENTS USED DURING INVESTIGATION

A. Brinell Hardness Tester

Model: King Portable Brinell
S/N: 4815
Calibration Date: 8/29/79

Test Blocks: Detroit Testing Machine Co.
Test Condition: Load 3000 kg
Hultgren Ball

	Test Block	Reading
No. 1	2.85 mm 461 BHN	2.87 mm 455 BHN
No. 2	3.70 mm 269 BHN	3.70 mm 269 BHN
No. 3	4.60 mm 170 BHN	4.60 mm 170 BHN

B. Rockwell Hardness Tester

Make & Model: Clark Twintester
S/N: T785
Calibration date: 8/6/79

Test Block: S/N 780128
Hardness Rkw C62.8 ± 0.5

	Readings (Rkw C)
	62.5
	63.2
	63.3
	63.3
	63.2
Ave:	63.1

APPENDIX A (Continued)

B. Rockwell Hardness Tester (Continued)

Test Block: S/N 786095
Hardness Rkw C23.0 ± 1.0

	Readings (Rkw C)
	23.5
	23.6
	23.4
	23.6
	23.5
Ave:	23.5

Test Block: S/N 77248
Hardness Rkw B88.9 ± 1.0

	Readings (Rkw B)
	88.6
	89.1
	88.7
	89.3
	89.0
Ave:	88.9

C. Equotip Hardness Tester

S/N: 23.341

Primary Standard Test Block supplied by manufacturer.
L = 839 ± 6

Date	Readings
8/10/79	842
	841
	841
	844
8/13/79	840
	839
	844
9/12/79 (on site)	839
	841
	842

APPENDIX A (Continued)

C. Equotip Hardness Tester (Continued)

Date	Readings
9/13/79 (on site)	834
	835
	839
	838
9/14/79 (on site)	841
	845
	843
	841
	840

Secondary Test Block developed by Technimet (See Appendix B for Description and Test Results).
Nominal L= 402-421

Date	Readings
9/12/79 (on site)	411
	412
	415
	413
9/13/79 (on site)	407
	418
	408
	410
9/14/79 (on site)	408
	407
	410
	413

D. Portable Metallograph

Make & Model: Unitron Rollscope Series DMR
S/N: RMM 1325
Objective Lens: Unitron MPL 10X
Eyepiece Lens: Unitron WFH10XR
Camera: Polaroid
Film Type: 3¼" x 4¼" Polaroid Type 667

Calibration Date: 9/3/79
Nominal Magnification: 100X
Actual Magnification using Stage Micrometer: 104.3X (See Fig. A-1)

APPENDIX A (Continued)

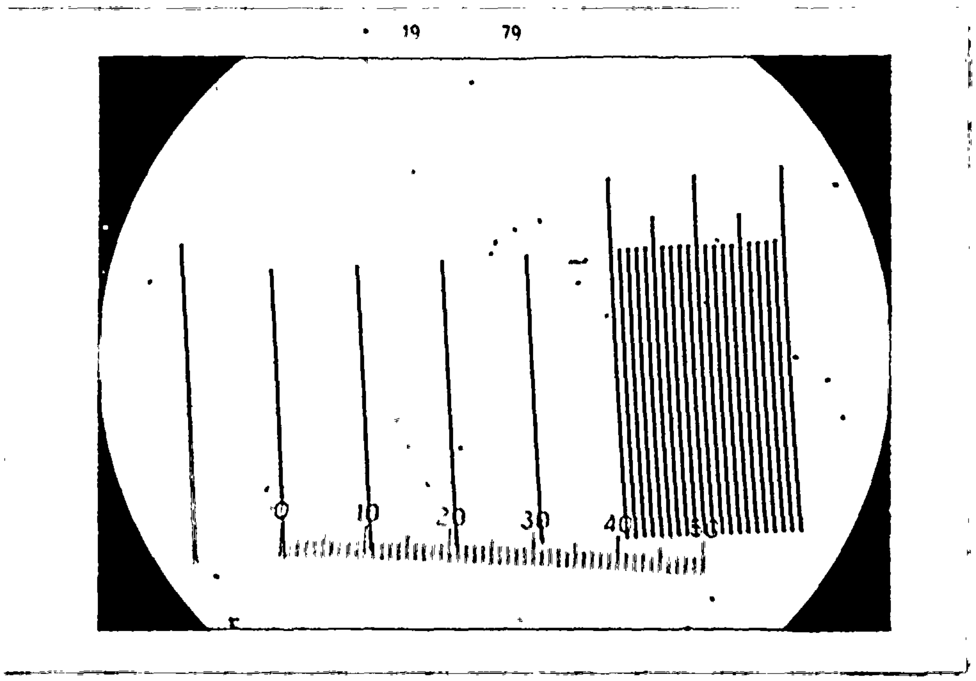
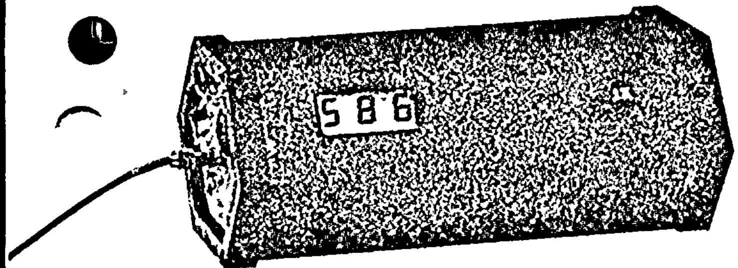


Fig. A-1 - Stage Micrometer measured with Rollscope. Nominal magnification = 100X. Actual magnification = 104.3X.

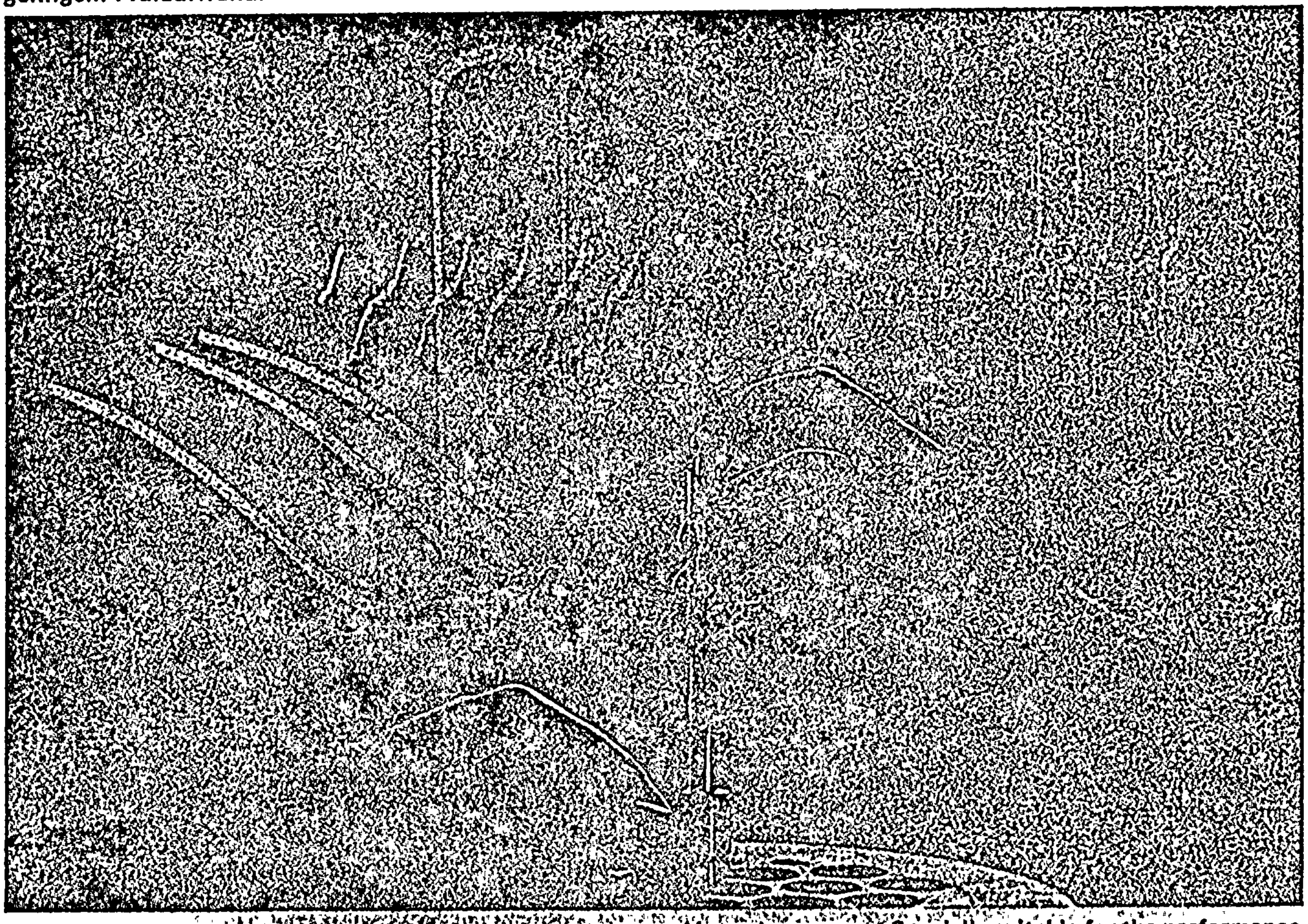
Härtemessgerät
Duromètre
Hardness Tester



EQUOTIP ist ein besonders leichtes, tragbares Härtemessgerät für metallische Werkstoffe, nach neuartigem, dynamischen Schnell-Prüfverfahren (EQUO-Verfahren).
Charakteristik: grosser Messbereich und hohe Genauigkeit, verbunden mit einfachster Handhabung und geringem Prüfaufwand.

EQUOTIP est un instrument de mesure portatif spécialement léger pour contrôler la dureté de matériaux métalliques, d'après un procédé dynamique nouveau et rapide (procédé EQUO).
Caractéristiques: grande portée de mesure, haute exactitude et d'un maniement simple.

EQUOTIP is an extremely lightweight, portable hardness measuring instrument for metallic materials, utilizing a novel, dynamic high-speed hardness testing procedure (EQUO method).
It is characterized by a large measuring range and extreme accuracy, combined with most simple handling and uncomplicated testing procedure.



Speziell geeignet für die rasche und aussagekräftige Härtemessung:

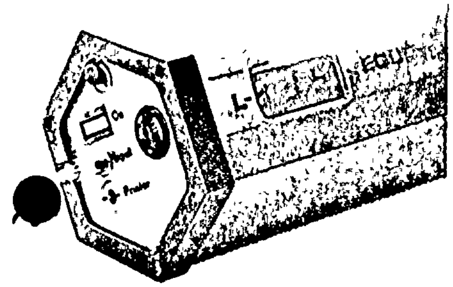
- an Ort und Stelle an schweren und grossen Werkstücken oder festverlegten Anlageteilen
- während der Produktion, insbesondere an Serienteilen
- an bereits montierten Maschinen
- im Materiallager zur Werkstoffidentifikation
- an schwer zugänglichen Stellen und bei knappen Platzverhältnissen.
- zur Untersuchung des Härteverlaufes über grössere Werkstückbereiche

Spécialement approprié pour la mesure rapide et sûre de la dureté:

- sur place pour des pièces grandes et lourdes ou pour des parties d'installation ne pouvant être déplacées,
- pendant la production, surtout pour des pièces de série,
- pour des machines déjà installées,
- pour identifier la dureté du matériau dans les lieux de stockage,
- à des endroits difficilement accessibles ou exigus,
- pour examiner l'uniformité de la dureté sur des pièces de grandes dimensions.

Specially suitable for the performance of rapid and reliable hardness tests:

- on site testing of large and heavy workpieces or fixed installation components,
- during production, especially during mass production,
- for machines already in place,
- at material storage depots for the identification of materials,
- at difficultly accessible locations and in cramped space conditions,
- for testing variations in hardness of especially large workpieces.

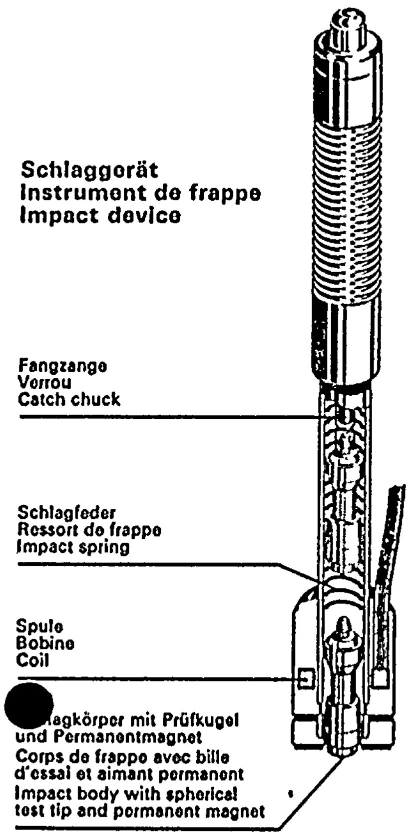


Das neuartige Messprinzip (EQUO-Verfahren)

Bei der Prüfung wird ein Schlagkörper mit einer Hartmetall-Prüfkugel durch Federkraft gegen die Prüffläche geschlagen und prallt dann wieder zurück. Die Aufprall- und Rückprallgeschwindigkeiten werden gemessen. Dies geschieht so, dass ein im Schlagkörper eingebauter Permanentmagnet während des Prüfschlages eine Spule durchfährt und im Vor- und Rückwärtsweg elektrische Spannungen induziert, welche sich proportional zu den Geschwindigkeiten verhalten. Die Messwerte aus Aufprall- und Rückprallgeschwindigkeit werden im Anzeigegerät zum Härtewert L verarbeitet.

Der Härtewert «L»

Dieser für die Härtemesstechnik neue Begriff ist der mit 1000 multiplizierte Quotient aus Rückprall- und Aufprallgeschwindigkeit des Schlagkörpers. Bei härterem Material ergibt sich eine grössere Rückprallgeschwindigkeit als bei Material von geringer Härte. Demnach ist der Härtewert L umso grösser, je härter das geprüfte Material ist. Neben der Härte des Werkstoffes geht auch dessen Elastizitätsmodul in die Messung ein. Der Einfluss des E-Moduls zeigt sich darin, dass bei Werkstoffen gleicher Härte, aber verschieden grossem E-Modul, der Werkstoff mit dem kleineren E-Modul einen grösseren L-Wert ergibt. Unter Bezugnahme auf eine bestimmte Werkstoffgruppe (z. B. Stähle, Aluminium etc.) bildet der L-Wert ein direktes Härtemass und wird als solches verwendet.
Für den Vergleich mit eingeführten statischen Härtewerten (Brinell, Vickers, Rockwell C) wurden Vergleichskurven erarbeitet, womit L-Werte in die entsprechenden Werte dieser Verfahren umgewandelt werden können.
Der L-Wert steht für LEEB-Wert, benannt nach dem Erfinder des Messverfahrens, Dipl.-Ing. Dietmar Leeb.



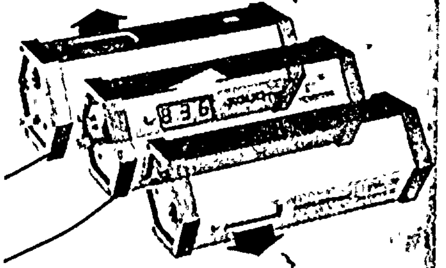
Principe de la nouvelle méthode de mesure (procédé EQUO)

Pendant l'épreuve un corps de frappe constitué par une bille de carbure fritté est lancé par la force élastique d'un ressort sur la surface à examiner et rebondit. La vitesse d'impact et la vitesse de rebondissement du corps de frappe sont mesurées. Un aimant permanent monté dans le corps de frappe traverse avant et après l'impact une bobine qui produit des tensions électriques induites proportionnelles aux vitesses. Les valeurs mesurées des vitesses d'impact et de rebondissement sont transformées dans l'instrument indicateur en valeurs de dureté L.

Valeur de dureté «L»

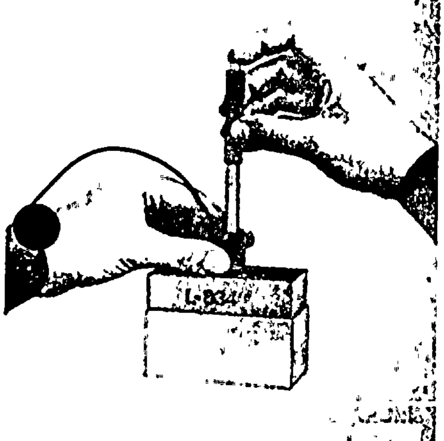
Cette nouvelle désignation dans la technique de la mesure de la dureté est égale au quotient de la vitesse de rebondissement par la vitesse d'impact du corps de frappe, multiplié par 1000. Pour un matériau plus dur la vitesse de rebondissement est plus grande que pour un matériau moins dur. La valeur de dureté L est donc d'autant plus grande que le matériau éprouvé est plus dur. A côté de la dureté du matériau entre aussi dans la mesure son module d'élasticité. L'influence du module d'élasticité est mise en évidence par le fait que pour deux matériaux ayant la même dureté, mais des modules d'élasticité différents, la valeur de dureté L est plus grande pour le matériau ayant un module d'élasticité plus petit. Si l'on considère un groupe de matériaux déterminé (p. e. aciers, aluminium etc.) la valeur de dureté L est une valeur de dureté directe et utilisée comme telle.
Pour comparer avec les valeurs de dureté statiques traditionnelles (Brinell, Vickers, Rockwell C) des courbes ont été établies à l'aide desquelles les valeurs L peuvent être transformées en valeurs correspondantes de ces procédés.
Valeur L signifie valeur LEEB, du nom de l'inventeur de ce procédé, l'ingénieur diplômé Dietmar Leeb.

Ablesung
Lecture
Reading



The new measuring method (EQUO method)

During hardness testing an impact body possessing a test tip formed of tungsten carbide is caused to strike under spring force against the test surface and rebounds therefrom. The impact and rebound velocities are measured. This is accomplished by a permanent magnet mounted in the impact body which moves through a coil during the test impact and, during the forward and return movement, induces electrical voltages which are proportional to these velocities. The measurement values derived from the impact and rebound velocities are processed at the indicator device into hardness values L.



The hardness value «L»

This new expression for the hardness measuring procedure constitutes the quotient of the rebound and impact velocities of the impact body multiplied by 1000. With harder materials there is a larger rebound velocity than in the case of materials of lesser hardness. Hence the hardness value L increases proportionally with the hardness of the material tested. Apart from the hardness of the material its modulus of elasticity is also introduced into the measurement. The affect of the E-modulus is such that in the case of materials of the same hardness but different E-modulus, the material with the smaller E-modulus furnishes a larger L-value. When related to a certain group of materials (e.g. steel, aluminium etc.) the L-value constitutes a direct hardness measurement and is used as such.
There have been derived conversion curves for comparison with standard static hardness values (Brinell, Vickers, Rockwell C), and thus L-values can be converted into the corresponding values of such hardness testing procedures.
L-value means the LEEB-value, according to the name of the inventor of the measuring procedure Dipl.-Ing. Dietmar Leeb.

Technische Daten

Messbereich: (E-Modul 210000 N/mm²)

Bereich des L-Wertes	äquivalente statische Härte
300–700	80–440 Brinell (30 D²)
300–880	80–940 Vickers
510–880	20– 68 Rockwell C

für Aluminium-Gusslegierungen (E-Modul 65000 bis 85000 N/mm²)

Bereich des L-Wertes	äquivalente statische Härte
200–560	30–160 Brinell (10 D²)

max. zulässige Härte des Prüflings 940 HV bzw. 68 HRC

Messgenauigkeit: mittlere Messunsicherheit ±0,8% (bezogen auf L=800) bzw. ± 6 L-Einheiten

Schlagkörper: Masse m=5,5 g, Schlagenergie E=11,0 Nmm (0,011 kgm²sec⁻²)
Prüfkugeldurchmesser d=3,0 mm, Material der Prüfkugel: Hartmetall mit ca. 1600 HV

Anzeigegerät: Speisung durch 3 Monozellen à 1,5 V, Betriebsdauer mit einem Batteriesatz bei 20°C ca. 50 Stunden
zulässiger Temperaturbereich: +5°C bis +50°C, Printeranschluss vorhanden

Masse und Gewichte: Schlaggerät ⌀ 20 mm, Länge 150 mm, Gewicht ca. 75 g, Anzeigegerät 245×112 mm, Gewicht ca. 900 g (mit Batterien), Tragkoffer mit Normalzubehör 325×250×185 mm, Gewicht ca. 6800 g

Données techniques

Portée de mesure: pour aciers (module d'élasticité 210000 N/mm²)

portée de la valeur L	dureté statique équivalente
300–700	80–440 Brinell (30 D²)
300–880	80–940 Vickers
510–880	20– 68 Rockwell C

pour alliages d'aluminium pour fonderie (module d'élasticité 65000–85000 N/mm²)

portée de la valeur L	dureté statique équivalente
200–560	30–160 Brinell (10 D²)

dureté max. admissible de la pièce à contrôler 940 HV respectivement 68 HRC

Exactitude de mesure: incertitude moyenne de mesure ±0,8% (relativement à L=800) respectivement ±6 unités L

Corps de frappe: masse m=5,5 g, énergie d'impact E=11,0 Nmm (0,011 kgm²sec⁻²)
diamètre de la bille d'épreuve d=3,0 mm, qualité de la bille: carbure fritté d'une dureté d'environ 1600 HV

Instrument indicateur: alimentation par 3 monocellules à 1,5 V, durée de fonctionnement avec 1 jeu de piles à 20°C environ 50 heures, température admissible: de +5°C à +50°C, équipé d'une prise pour enregistreur

Dimensions et poids: Instrument de frappe ⌀ 20 mm, longueur 150 mm, poids environ 75 g, instrument indicateur 245×112 mm, poids environ 900 g (y compris piles), coffret avec instruments 325×250×185 mm, poids environ 6800 g

Technical data

Measuring range: for steel (E-modulus 210000 N/mm²)

measuring range of L-value	equivalent static hardness
300–700	80–440 Brinell (30 D²)
300–880	80–940 Vickers
510–880	20– 68 Rockwell C

for cast aluminium alloys (E-modulus 65000–85000 N/mm²)

measuring range of L-value	equivalent static hardness
200–560	30–160 Brinell (10 D²)

maximum permissible hardness of the sample 940 HV or 68 HRC respectively

Measuring accuracy: average measurement deviation ±0.8% (related to L=800) or ±6 L-units respectively

Impact body: mass m=5.5 gms, impact energy E=11.0 Nmm (0.011 kgm²sec⁻²)
diameter of the spherical test tip d=3.0 mm, material of the spherical test tip: tungsten carbide of approximately 1600 HV

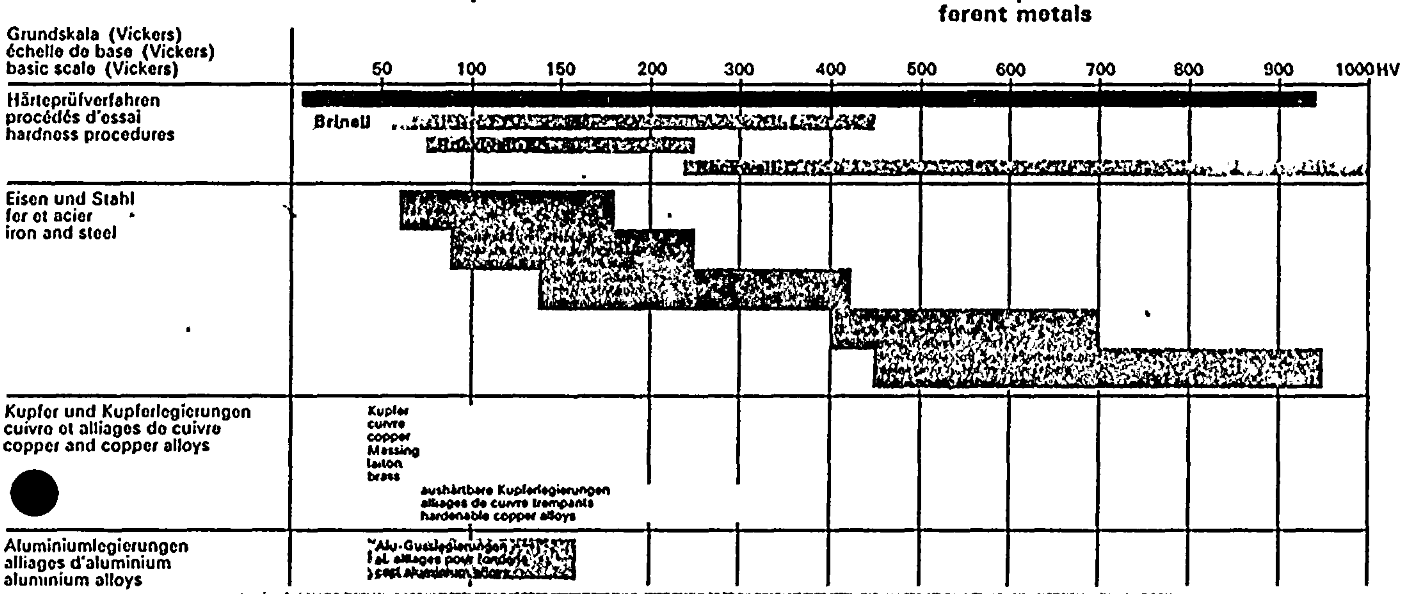
Indicator device: power supplied by 3 single battery cells each 1.5 volts, operation duration with a battery set of about 50 hours at 20°C, permissible temperature range: +5°C to +50°C, equipped with printer terminal

Dimensions and weights: impact device ⌀ 20 mm, length 150 mm, weight approximately 75 gms, indicator device 245×112 mm, weight approximately 900 gms (inclusive of batteries), carrying case with standard equipment 325×250×185 mm, weight approximately 6800 gms.

Messbereiche der wichtigsten Härteprüfverfahren im Vergleich zur Härte verschiedener Metalle

Portées de mesure des procédés principaux pour l'essai de dureté en comparaison avec la dureté des métaux

Measuring ranges of the most important hardness measuring procedures in comparison to the hardness of different metals



Patente angemeldet
Technische Änderungen vorbehalten
Made in Switzerland

Brevets déposés
Modifications techniques réservées
Made in Switzerland

Patents applied for
Subject to changes
Made in Switzerland

76 11 299

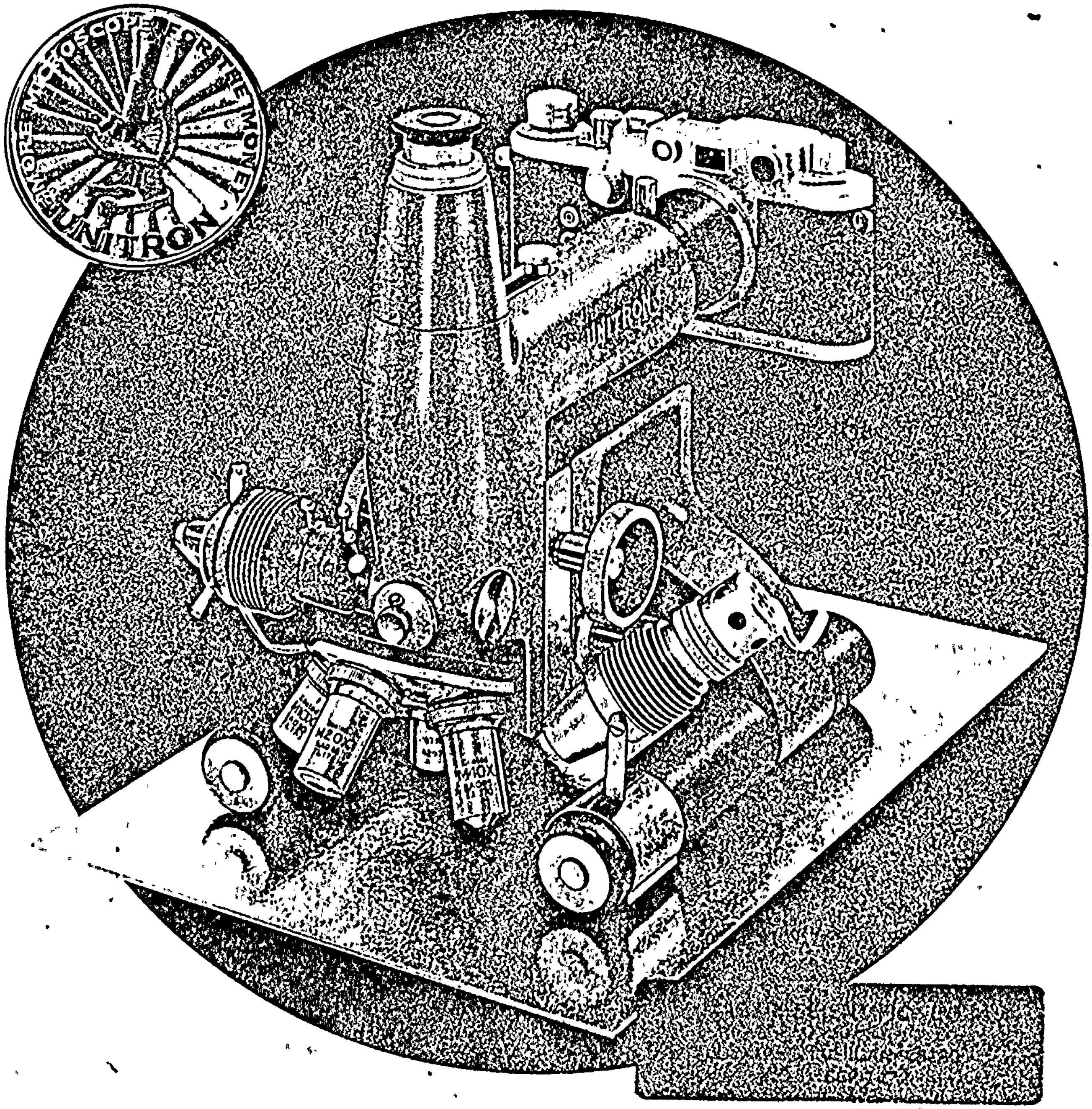
PROCEQ SA
Riesbachstrasse 57
CH-8034 Zürich

Tel. 01 47 78 00
Telegr. procequipment zurich
Telex 53 357 proce ch

UNITRON

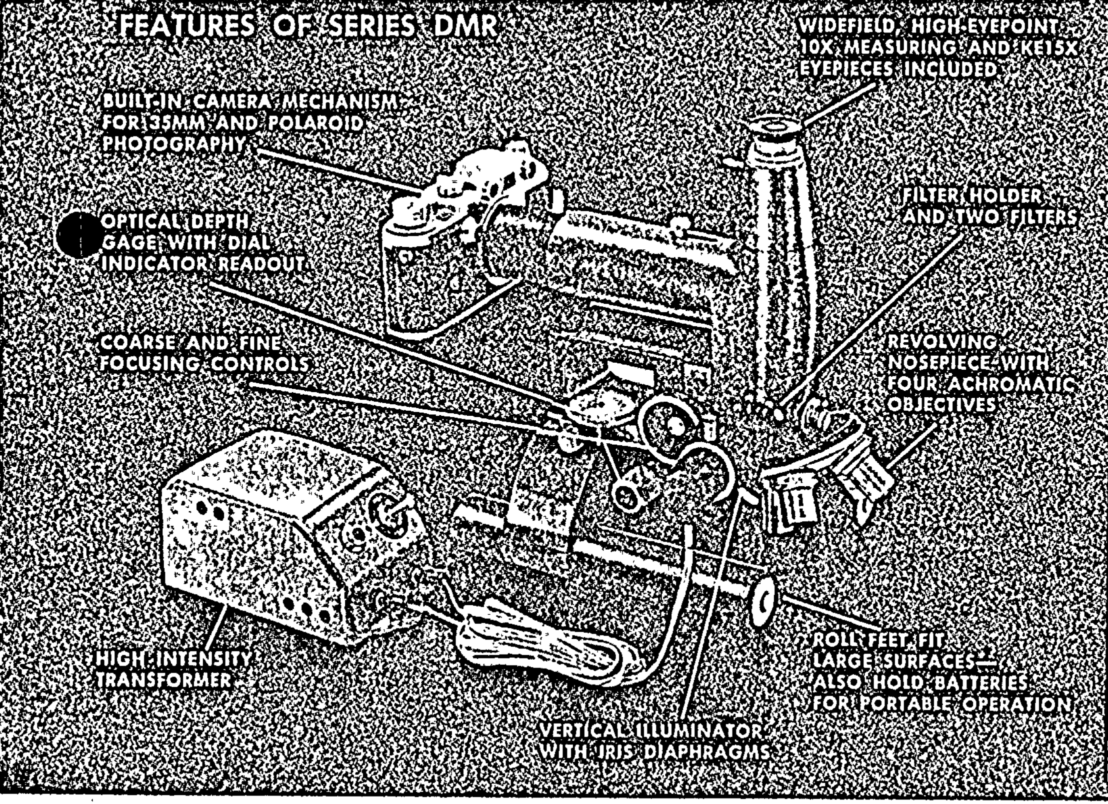
SERIES DMR

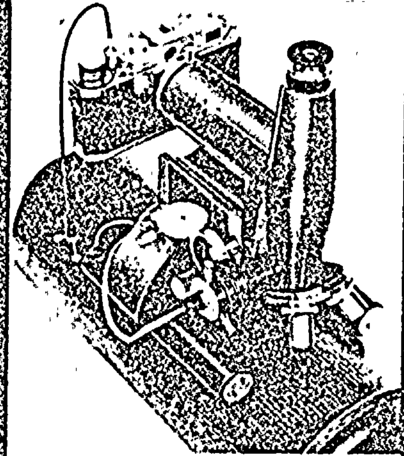
PORTABLE DEPTH—MEASURING ROLLSCOPE



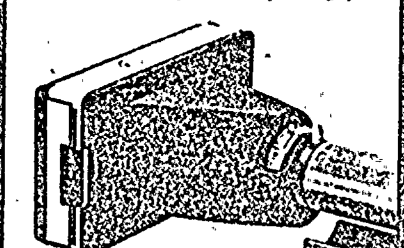
- **FOR ALL TYPES OF METAL SURFACES**
  Examine grain structure and check surface finish on workpieces of all sizes, flat areas or curved . . . on objects too large to place on a microscope stage.
- **ON-THE-SPOT EXAMINATION**
  Lightweight and fully portable . . . operates from self-contained batteries or transformer. Permits non-destructive metallographic tests of turbine shafts, boiler walls, steel plate, welded joints, printing rollers, etc.
- **MEASURES IN 3-DIMENSIONS**
  Measure depth of pits, etch, etc. using optical depth indicator . . . make linear measurements using micrometer eyepiece.
- **COMPLETE MAGNIFICATION RANGE**
  Powers from 50-600X with standard optics . . . widefield eyepieces give 50% larger field of view.
- **BUILT-IN CAMERA MECHANISM**
  Photograph with standard-equipment 35mm or accessory Polaroid Land camera attachments.

UNITRON SCIENTIFIC, INC. • 66 NEEDHAM STREET • NEWTON HIGHLANDS, MA. 02161





Measuring etch depth on a printing plate.



Accessory Polaroid Land camera attachment.

SPECIFICATIONS and PRICE LIST for SERIES DMR

Portable stand with coarse and fine focusing; quadruple revolving nosepiece with four coated, achromatic, parfocal objectives: M5X(N.A. 0.10), M10X(N.A. 0.30), M20X(N.A. 0.40), M40X(N.A. 0.65); two coated eyepieces: widefield, high-eyepoint WFH10XR Micrometer with linear reticle and Ke15X; vertical illuminating system with field and aperture diaphragms; filter slot with green and yellow filters; built-in camera mechanism including 35mm camera coupling tube with 4X photo-lens for Leica-type camera back (back not included), photo-finder reticle; bulb cord assemblies for both battery and transformer operation; transformer 115 for volts A.C. with two-intensity switch; set of batteries; spare bulbs; fitted wooden carrying case and plastic dustcover.

DMRE: with optical depth gage, 0.0001" graduations .... $

DMRM: with optical depth gage, 0.002mm graduations .... $

DMRA: as above, but without optical depth gage feature .... $

Models above but including 35mm Camera Back, add .... $

ADDITIONAL ACCESSORIES for SERIES DMR INCLUDE—

- POLAROID LAND CAMERA ATTACHMENT: Model PB, complete including camera back; for "10 second" photography ...... $
- OBLIQUE ILLUMINATOR: Model DMR-OL, attaches to microscope foot as illustrated on reverse page; uses standard bulb cord ... $
- M3X OBJECTIVE: for 30X, at field diameter of 6mm .... $
- M60X OBJECTIVE: for powers of 900X, 1200. (N.A. 0.80) .... $
- WF20X EYEPIECE: widefield type .... $
- INTERFERENCE ACCESSORIES: Model DMR-INT, complete with 10X Interference Objective, plane proof plate, monochromatic filter. (Request Bulletin INT for further details.) .... $
- POLARIZING ACCESSORIES: DMR-PO, polarizer and analyzer . $
- FILAR EYEPIECE TUBE: Model DMR-FT, required when using filar eyepieces; interchanges with standard tube .... $
- FILAR MICROMETER EYEPIECE: choice of inch or metric type .. $

APPENDIX B

Secondary Test Blocks:

Since hardness readings obtained with the Equotip tester depend on the materials' elastic modules, the brass test block S/N 77248 is not desirable for calibration when the subject material is steel. Consequently, we developed 3 secondary test blocks from 8620H alloy steel to cover the expected hardness range.

These test blocks were thermally treated for uniformity and tested with the Clark Twintester after careful surface preparation. The hardnesses obtained compared well with Equotip "L" values after both were converted to DPH as shown in Table B-1. Tables B-2 through B-4 list the individual hardness readings.

APPENDIX B (Continued)

TABLE B-1

HARDNESS COMPARISONS
August 10-13, 1979

	DPH-Clark	Equotip Vert-Up	Equotip Horiz.	Equotip Vert. Dwn.
Test Block #1				
Average	161	155	154	148
Range	158-165	149-156	147-158	144-154
Test Block #2				
Average	198	194	194	190
Range	190-203	190-195	188-199	189-195
Test Block #3				
Average	277	280	281	276
Range	272-283	278-287	274-289	271-286

DPH Clark - Converted from Rockwell
DPH Equotip - Converted from "L"

APPENDIX B (Continued)

TABLE B-2

Secondary Standard Test Block #1

Material: AISI 8620H
Size: 2½" dia. x 1 3/4"

Thermal Treatment:

1750° F - 12.0 hrs.
Oil Quench
1200° F. - 21.0 hrs.
Air Cool

HARDNESS

	Clark	Equotip ↑	Equotip →	Equotip ↓
Rkw B	82.8	415	425	430
	82.4	411	419	433
	83.0	416	415	429
	81.5	413	422	425
	82.0	418	431	436
	82.0	414	423	425
	82.2	417	421	429
	81.6	413	421	424
	82.8	412	419	427
	83.4	408	423	425
POSITION CORR.	---	---	-10	-22 *
CORR. AVE.	82.4	413.7	411.9	406.3
CORR. RANGE	81.5-83.4	408-418	405-421	402-414
CORR. AVE. EQUIV. DPH	161	155	154	148
CORR. RANGE EQUIV. DPH	158-165	149-156	147-158	144-154

* Refer to Page B-7, Table 5 for the position correction L-values (up, down, horizontal) for the Equotip Hardness Comparisons.

APPENDIX B (Continued)

TABLE B-3

Secondary Standard Test Block #2

Material: AISI 8620H
Size: 1 3/4" dia. x 3 1/2"

Thermal Treatment:

1750° F. - 12.0 hrs.
Oil Quench
1200° F. - 5.0 hrs.

HARDNESS

	Clark	Equotip↑	Equotip→	Equotip↓
Rkw B	91.0	465	471	479
	91.5	465	467	482
	91.0	460	473	481
	91.7	463	473	480
	92.0	461	471	479
	92.0	460	469	477
	90.5	462	479	480
	90.6	461	474	486
	92.0	465	472	484
	89.5	464	471	477
	90.3			
POSITION CORR.	---	---	-10	-21
CORR. AVE.	91.1	462.5	462.0	459.5
CORR. RANGE	89.5-92.0	460-465	457-469	458-465
CORR. AVE. EQUIV. DPH	198	194	194	190
CORR. RANGE EQUIV. DPH	190-203	190-195	188-199	189-195

APPENDIX B (Continued)

TABLE B-4

Secondary Standard Test Block #3

Material: AISI 8620H
Size: 2½" dia. x 3½"

Thermal Treatment:

1750° F. - 12.0 hrs.
Oil Quench
350° F. Temper - 4.0 hrs.
Air Cool

HARDNESS

	Clark	Equotip ↑	Equotip →	Equotip ↓
Rkw C	26.0	549	555	569
	26.3	557	557	566
	26.0	552	563	565
	27.0	551	559	563
	26.0	548	558	563
	27.5	548	560	563
	27.5	551	568	575
	26.8	548	563	570
	27.0	551	558	568
	26.5	554	563	561
POSITION CORR.	---	---	-10	-19
CORR. AVE.	26.7	550.9	550.4	547
CORR. RANGE	26.0-27.5	548-557	545-558	542-556
CORR. AVE. EQUIV. DPH	277	280	281	276
CORR. RANGE EQUIV. DPH	272-283	278-287	274-289	271-286

9. Conversion Values between L-Value and Static Hardness Numbers

As an appendix to these Operating Instructions there are attached curves and tables providing conversion values between the hardness value L and the static indentation hardness values HB, HV and HRC

In order to produce these conversion curves hardness tests were carried out according to all four hardness testing procedures upon a large number of steel and aluminum samples as well as upon standard test blocks to be used for hardness testing machines. Then, the obtained measurement values were processed with the method of least squares (non-linear regression) into the individual conversion-curves

9.1 Validity of the Values of the Curves and Tables

Values for steel and cast steel (E-modulus approximately 210000 N/mm²)
The values are valid for unalloyed and low alloy steels and cast steel in hot rolled and thermally treated condition.
For the Brinell hardness the test was carried out with the load factor $F = 30D^2$.
F = Test load
D = Ball diameter
Values for cast aluminum alloys (E-modulus 65000–85000 N/mm²)
For the determination of the conversion curves there were employed the following cast alloys in a non-heat-treated and a quenched and tempered condition:

Abbreviation according to VSM 10895		Abbreviation according to DIN 1725
G-AlSi11	corresponds approx.	G-AlSi12
G-AlSi9Mg	corresponds approx.	G-AlSi10Mg
G-AlSi7MgTi	corresponds approx.	G-AlSi5Mg
G-AlCu5Ti	corresponds approx.	G-AlCu4Ti
G-AlCu5MgTi	corresponds approx.	G-AlCu4TiMg

The Brinell test was carried out with the load factor $F = 10D^2$.
The conversion values are also approximately valid for malleable aluminum alloys. For some alloys, such as for instance Al-Mn and AlCu4Mg1 ? there must be reckoned with larger deviations.

9.2 Deviations from the Conversion Values occur in the following situations:

- With high alloy and/or cold worked steels as well as austenitic steels
- Drawn and also in part rolled steels frequently lead to too large L-values due to the pronounced cold worked regions near to the surface and thus simulate too great [illegible] [illegible] steels [illegible] should be tested over their cross-sect[illegible]
- Surface hardened especially however casehardened steels produce too low L-values because of their soft core.
- With high-speed steels, hot work steels and ledeburitic chromium steels (carbide rich group of cold work steels) a local increase of the E-modulus is caused by the hard substances embedded in the matrix (ledeburitic tungsten carbide e.g of the type M7C3 and M6C). Thereby too small L-values are produced
- When testing magnetic materials the velocity transmitter in the impact body is briefly influenced by their magnetic field, so that, slight deviations can exist in the measured L-value

9.3 Inaccuracies caused by Conversion

The conversion of mutual hardness values is basically associated with inaccuracies. This is not only so for the conversion of the L-values into static indentation hardness numbers, but also for the conversion of the static indentation hardness numbers among themselves. Thus, during the conversion, the measurement deviations of both hardness testing procedures can add to one another. Additionally, there is no clear physical correlation which exists between the individual hardness testing procedures.

9.4 Producing Conversion Curves at the Plant

The accuracy during conversion can be improved if, for those types of materials which are frequently utilized, individual curves are worked out at the plant or workshop.
The types of material, apart from being grouped in terms of their E-modulus, also can be grouped together with respect to their yield point or alloy type.
When producing conversion curves the following points are to be observed.

- The sample surface must be prepared extremely carefully
- The sample dimensions, if possible, should be chosen to be so large that no coupling is needed.
- The correct read-out of the EQUOTIP hardness tester is to be checked for each serie of measurements at the hardness standard test block.
- The function of the static hardness testing machine and the correct optical evaluation of the indentations is to be checked for each measurement serie with the help of the standard test block of the corresponding measuring range.

Steel and Cast Steel (E-Modulus 210000 N/mm²) — Table 5

Conversion Table L-value – Brinell Hardness (Impact Device D)

Validity of Table:
- for steel and cast steel types as listed in Section 9.1
- for vertical impact directions (from top towards bottom)

L-value	Brinell Hardness HB (F = 30 D²)	Conversion-Deviation* ±HB
300	79	
305	82	
310	85	
315	88	
320	91	
325	93	
330	96	
335	99	
340	103	
345	106	
350	109	
355	112	10
360	115	
365	119	
370	122	
375	125	
380	129	
385	132	
390	136	
395	140	
400	143	
405	147	
410	151	
415	154	
420	158	
425	162	
430	166	
435	170	
440	174	
445	178	
450	182	
455	186	
460	190	
465	195	
470	199	13
475	203	
480	208	
485	212	
490	217	
495	221	
500	226	
505	230	
510	235	
515	239	
520	244	
525	249	
530	254	
535	259	
540	263	16
545	268	

L-value	Brinell Hardness HB (F = 30 D²)	Conversion-Deviation* ±HB
550	273	
555	278	
560	283	
565	289	
570	294	
575	299	
580	304	
585	309	16
590	315	
595	320	
600	325	
605	331	
610	338	
615	342	
620	347	
625	353	
630	359	
635	364	
640	370	
645	376	
650	382	
655	388	
660	393	20
665	399	
670	405	
675	411	
680	417	
685	424	
690	430	
695	436	
700	442	

Corrections for other impact directions

Measured L-value	Subtract from measured L-value (horizontal)	Subtract from measured L-value (vertical upwards)
300		24
400		22
500	10	20
600		18
700		19

Intermediately located impact directions should be linearly interpolated

* Deviation of the Table value in comparison to direct Brinell Hardness testing

Steel and Cast Steel (E-Modulus 210000 N/mm²) — Table 6

Conversion Table L-value – Vickers Hardness (Impact Device D)

Validity of Table:
- for steel and cast steel types as listed in Section 9.1
- for vertical impact direction (from top towards bottom)

L-value	Vickers Hardness HV	Conversion-Deviation* ±HV
300	79	
305	82	
310	85	
315	88	
320	91	
325	93	
330	96	
335	99	
340	103	
345	106	
350	109	
355	112	10
360	115	
365	119	
370	122	
375	125	
380	129	
385	132	
390	136	
395	140	
400	143	
405	147	
410	151	
415	154	
420	158	
425	162	
430	166	
435	170	
440	175	
445	179	
450	183	
455	187	
460	192	
465	196	13
470	201	
475	206	
480	210	
485	214	
490	219	
495	224	
500	229	
505	234	
510	239	
515	244	
520	249	
525	254	
530	259	16
535	264	
540	269	

L-value	Vickers Hardness HV	Conversion-Deviation* ±HV
545	274	
550	280	
555	285	
560	291	
565	296	
570	302	
575	307	16
580	313	
585	319	
590	325	
595	330	
600	336	
605	342	
610	345	
615	354	
620	360	
625	366	
[illegible]	373	
635	379	
640	385	
645	392	
650	398	20
655	404	
660	411	
665	418	
670	424	
675	431	
680	438	
685	444	
690	451	
695	458	
700	465	
705	472	
710	480	
715	488	
720	496	
725	505	27
730	514	
735	524	
740	534	
745	544	
750	554	
755	565	
760	577	
765	588	
770	600	
775	612	40
780	624	

→ NRC-06

TECHNIMET CORPORATION

B-8

4379 SOUTH HOWELL AVENUE
MILWAUKEE, WISCONSIN 53207
414/483-0054

September 10, 1979

Mr. Richard Lofy
PARAMETER, INC.
Consulting Engineers
13545 Watertown Plank Rd.
Elm Grove, WI 53122

Dear Dick:

This will serve to summarize our progress and recommendations regarding NRC Task - 06:

A. Hardness Tests:

We have tested the Wilson portable M-9 tester along with a PROCEQ "Equotip" tester.

The M-9 measures directly in Rockwell B. The tester is clamped onto the pipe by a "motorcycle" type chain and the minor and major loads are applied through a built-in spring mechanism. The tester requires about 18" of clearance and weighs about 50 lbs., making it somewhat difficult to handle in some positions. It is also extremely sensitive to vibration and other sources of movement since hardness is measured by a dial indicator gage that senses 1 scale division (Rockwell point) for every .000080" vertical motion.

The Equotip tester works on a rebound principle and relates hardness to the inverse ratio impact and rebound velocity of a small impact body which is accelerated by a built-in spring mechanism. The measurements are completely electronic and are in "L" value which can be converted to more conventional tests such as Rockwell, Brinnell or Vickers.

The test is easy to perform, rapid and repeats well if due care is taken to prepare the surface. The impact body leaves an impression which is about half as deep as the Rockwell B test so this tester appears to be much less sensitive to surface than the standard "Shore" rebound test.

A. Hardness Tests: (Continued)

Since the impact body is accelerated by a spring and readings are taken electronically, the test can be taken in any position and the errors due to human judgement are eliminated.

The hardness results on the sample 24" dia. tube that Parameter procured are shown in Table A. The three methods of tests we used were (1) Brinell (Portable King Tester), (2) Rockwell B with M-9 tester and (3) Equotip. All 3 readings are very similar and would normally be considered within expected accuracy especially since conversions from one hardness scale to another are involved.

We further tested the Equotip tester by comparing the hardness readings on secondary standards developed by Technimet. We used secondary standards because no low hardness steel test blocks were available. The results are summarized in Table B and indicate the Equotip has good precision and accuracy in the anticipated hardness range (Rkw B 75 - Rkw C 27).

These tests lead us to conclude the Equotip tester would provide the following advantages over the M-9 tester:

1. Substantially greater accessibility. The smaller and highly portable probe will reach into tighter spaces and we could more accurately locate the point of test. With the M-9 there will probably be areas near the weld itself which cannot be tested due to interference with the anvil on the tester and the need to maintain perpendicularity.

2. More readings per location. We recommend at least 10 readings per location to assure accuracy. With the M-9 we would probably not be able to get more than 3 without exceeding the projected time schedule.

A. Hardness Tests: (Continued)

3. Greater hardness range. The M-9 tester requires a relatively complicated tear down and reassembly with a different load spring if hardness exceeds Rkw B-100. The Equotip requires no changes.

4. Simpler mode of operation.

5. Not as sensitive to vibration or outside forces which lead to large errors with the Rockwell test. Any influence of vibration can be minimized by averaging a large number of readings.

Mr. Joe Collins of the NRC agreed to the use of the Equotip tester for this project during our conversation with him Friday, September 7, 1979.

B. Grain Size:

The Unitron "Rollscope" portable metallograph was received, assembled and calibrated with a stage micrometer. The optics are within ± 5% specified accuracy and the unit measures 104.3X at the nominal 100X used for grain size determination.

Our recommended polishing procedure consists of rough grinding with 80grit Si C followed by 120, 240 and 400grit. This is followed by rough polishing with 6 micron diamond paste and final polish with 1 micron diamond on nylon cloth.

Both 3" diameter and 1" PSA backed grinding and polishing cloths on a rubber wheel in a 3/8" variable speed drill were found to be successful in obtaining a reasonable metallurgical polish suitable for grain size determination after a 2% nital etch. Photo-micrographs of reasonable quality were obtained.

As far as actual grain size determinations are concerned we recommend that any structure exhibiting fine grain (ASTM E 112) 5 or finer) be judged by visual comparisons to standard charts.

September 10, 1979

Mr. Richard Lofy
PARAMETER, INC.

B. Grain Size: (Continued)

This method will provide an accuracy of ± 1 ASTM number and is generally accepted for quality control purposes. If grain size is coarser than 5, we recommend using the circular intercept method.

It must be kept in mind that metallographic grain size in low carbon ferritic steels refers to ferrite grain size, not prior austenite grain size. Special fracture or carburizing tests are required to determine prior austenite grain size unless the structure is martensitic.

Mr. Collins concurred with the above recommendations regarding grain size evaluation by phone Friday, September 7, 1979.

Please call if you have any questions.

Thank you!

Very truly yours,

TECHNIMET CORPORATION

Manfred E. Suess

M. E. SUESS
President

MES/mp
Encl.

TABLE A

Hardness on 24" Tube Sample

Tester	Measurements
King Brinell	5.05 mm - BHN 140 5.05 mm - BHN 140
Wilson M-9 Vertical Up	Average of 7 Readings Rkw B 77.1 = BHN 137
Wilson M-9 Horizontal	Average of 3 Readings Rkw B 75 = BHN 133
Wilson M-9 Vertical Down	Average of 4 Readings Rkw B 73.7 = BHN 130
Equotip Vertical Up	Average of 15 Readings L-392.7 =BHN 138
Equotip Horizontal	Average of 15 Readings L-384.8 = BHN 132
Equotip Vertical Down	Average of 15 Readings L-389.8 = BHN 136

TABLE B

HARDNESS COMPARISONS
August 10-13, 1979

	DPH-Clark	EQUOTIP VERT-UP	EQUOTIP HORIZ	EQUOTIP VERT - DWN
Test Block #1				
Average	161	155	154	148
Range	153-165	149-156	147-158	144-154
Test Block #2				
Average	198	194	194	190
Range	190-203	190-195	188-199	189-195
Test Block #3				
Average	277	280	281	276
Range	272-283	278-287	274-289	271-286

DPH Clark - Converted from Rockwell

DPH EQUOTIP - Converted from "L"

APPENDIX C

Date of Tests: 8/29/79

24" dia. Tube. Full Scale Testing.

Hardness Check -
Hand Grind and Polish Surfaces.
60, 100, 240, 400 grit

Equotip

King Brinell*	Vert. Up	Horizontal	Vert. Down
	390	399	412
5.05 mm - 140 BHN	392	395	413
5.05 mm - 140 BHN	394	392	415
	393	386	413
	391	396	409
	391	386	408
	393	398	417
	390	388	408
	393	407	417
	401	399	406
	390	404	411
	393.	388	413
	392	399	410
	391	396	414
	396	390	412
TOTAL	5890	5923	6178
AVE.	392.7	394.8	411.8
POSITION CORRECTION	0	-10	-22
CORRECTED AVE.	392.7	384.8	389.8
BHN EQUIVALENT	138	132	136

*This instrument could be used because we had access to the I.D. of the tube.

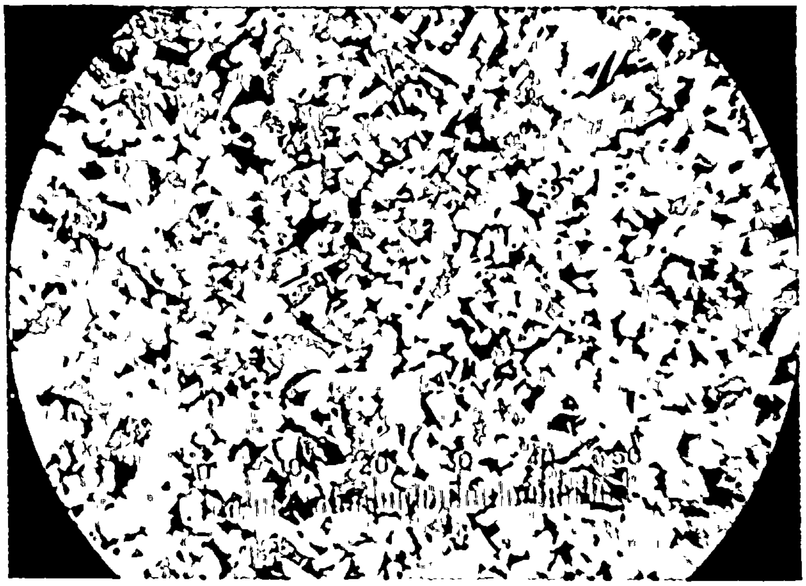


Fig. C-1 - Microstructure of 24 in. diameter tube supplied by Parameter, Inc.

Metallographic Procedure:

Hand grind with 3/8" electric drill

SiC Grits	60
	100
	240
	400
	600

Polish with diamond paste on nylon

Rough Polish - 6 micron
Final Polish - 1 micron

Etchant: 2% Nital

REVISIONS TO PREVIOUSLY SUBMITTED

AAB QUESTIONS

Q. 312.009

A failure to ιeither of the motor operated dampers WMA-AD-51A-1 ι-AD-51B-1 (refer to Figure 9.4-1) in the event of ;-of-coolant accident would provide an emergency ; train bypass path. Describe briefly what additionatection, if any, will be provided to prevent potenιnfiltered inleakage via this pathway.

Response:

Failure of WM1A-1 or WMA-AD-51B-1 to close is discussed in 92.1. The alarm referenced in 9.4.1.2.1 has recently ddded to the design.

Q. 312.015

You indicated in your response to Item 312.005 of the first acceptance review that you will qualify protective coatings to the requirements of ANSI N101.4-1972. However, Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water Cooled Nuclear Power Plants", June 1973, endorses ANSI N101.4-1972 on the condition that ANSI N45.2-1971 be used in conjunction with it. Accordingly, indicate your intended degree of compliance with the recommendation of Regulatory Guide 1.54 in this regard.

Response:

The protective coatings applied on WNP-2, with the exception of NSSS equipment, is qualified to ANSI N101.4. Also, the protective coating contractor is required to have both a Quality Assurance program and procedures which meet the requirements of ANSI N45.2, "Quality Assurance Requirements for Nuclear Power Plants". This meets the requirements of Regulatory Guide 1.54.

The NSSS equipment was prime coated with inorganic zinc. Inorganic zinc coatings meet the requirements of ANSI N101.2, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities". The coatings were manufactured and applied in accordance with the manufacturers instructions. However, Regulatory Guide 1.54 was not in effect at the time the NSSS equipment was ordered and consequently not invoked.

Although the Quality Assurance requirements of Regulatory Guide 1.54 were not invoked, the standard acceptance criteria for nuclear coatings, normally imposed by GE were in effect for the NSSS equipment coatings.

RESPONSES TO

STRUCTURAL ENGINEERING BRANCH (SEB)

QUESTIONS 130.010 - 130.044

Q. 130.010
(3.3.2)

Your response to Item 130.001 is not satisfactory. The tornado load listed in Tables 3.8-15 and 3.8-16 of the FSAR is the combined effect of three separate loads generated by the Design Basis Tornado. Accordingly, describe your procedures to combine the tornado wind load, the tornado differential pressure load and the tornado missile load. Provide your procedures to evaluate the effective loads resulting from tornado generated missiles.

Response:

Please refer to revised 3.3.2.2 for the information requested.*

*Draft FSAR page change attached.

3.3.2.2 Determination of Forces on Structures

Design static pressures, drag coefficients and wind pressures are selected in accordance with Reference 3.3-1. The provisions for gust factors and variation of wind velocity, noted therein, are not applied for reasons discussed below.

The wind velocity may vary with the height of the structures but, for conservatism, the wind force due to tornado loadings is ~~are~~ applied as a uniform static load invariant with the height above grade.

The 360 mph total wind velocity occurs only in a localized area but is used in the design over the full height of the projected area of the structure. The total wind velocity of 360 mph is, in effect, the gust wind velocity since by definition, a tornadic gust wind velocity is a high localized wind velocity of very short duration. Therefore, no additional gust factor is applied.

A drag coefficient of 1.3 is used to determine the total combined average pressure on the outside of each structure, of which 0.8 is used on the windward face and 0.5 on the leeward face.

Except for the steel superstructure atop the refueling floor (discussed in 3.3.2.2.1), the reactor building remains sealed through the tornado event and a differential pressure of 3 psi across the exterior and interior is considered in the design. All other Seismic Category I structures are provided with adequate openings to relieve a differential pressure of 3 psi in 3 seconds or are designed to withstand an external pressure drop of 3 psi.

The procedure used to transform the Design Basis Tornado wind velocity into an effective pressure applied to exposed surfaces of structures is as described in Reference 3.3-1 and is summarized as follows:

a. Dynamic Wind Loading

The same procedure as that utilized to transform the basic Design Basis Wind velocity in 3.3.1.2 is used with the exception that the velocity and velocity pressure are assumed not to vary with height.

The equivalent uniform tornado wind velocity used on the structure due to a tangential component of 300 mph and a translational component of 60 mph is 360 mph. The pressure loads are calculated on the basis of a uniform 360 mph wind velocity.

(1) The dynamic pressure on the structure is:

$$q = 0.002558 \times (360)^2 = 331.5 \text{ psf}$$

(2) The applied static pressures are:

(a) Windward pressure on walls:

$$p = 0.8 \times 331.5 = 265 \text{ psf}$$

(b) Leeward suction on walls:

$$p = 0.5 \times 331.5 = 166 \text{ psf}$$

(c) Total design pressure on the structure is the sum of 265 psf and 166 psf, or 431 psf.

The roof decking and siding enclosing the steel superstructure atop the refueling floor of the reactor building is designed to blow off the steel frame at a maximum differential pressure of 75 psf as discussed in 3.3.2.2.1.

b. Differential Pressure Loading

The differential pressure loading is calculated using the following pressure - time function:

The differential pressure is assumed to vary from zero to 3 psi at a rate of 1 psi per second and then return to zero at 1 psi per second.

c. Tornado - Generated Missiles

The procedure used for transforming the tornado-generated missile loadings into effective static loads is described in 3.5.3.

Mathematical models for the Design Basis Tornado take into consideration the phase relationship between the wind load and the differential pressure effects.

The tornado load, W', in the load combinations in Tables 3.8-15 and 3.8-16 constitutes the combined effect of the three separate loads a, b and c generated by the Design Basis Tornado. The three loads are combined in the following manner to obtain the combined effect:

a. $W' = W_w$

b. $W' = W_p$

c. $W' = W_m$

d. $W' = W_w + W_p$

e. $W' = W_w + W_m$

f. $W' = W_w + W_p + W_m$

where:

W' = Total tornado load

W_w = Tornado wind load

W_p = Tornado differential pressure load

W_m = Tornado-generated missile load

3.3.2.2.1 Additional Design Features

The structural steel frame superstructure atop the refueling floor of the reactor building is designed to withstand the design basis tornado. However, all the siding and roof decking enclosing the steel superstructure is designed for

Q. 130.011
(3.3.2)
(3.8.2)
(372.8)

Airborne dust has been identified as a possible design basis meteorological parameter in Item 372.008 of our request for additional information. Describe your procedures to evaluate airborne dust loadings. Indicate whether an airborne dust load was included in any of the loads and load combinations defined in Section 3.8.2.3 of the FSAR.

Response:

Airborne dust loading is not considered in design of structures at WNP-2. Accumulation of dust to depths necessary to cause significant loading on structures has not been observed in the Hanford area. At WNP-2, structures are designed for a 20 psf snow load, which is considered to encompass dust loading due to any credible dust storm or long-term accumulation.

Q. 130.012
(3.5.3)

Indicate the formula you used to determine the penetration depth of missiles into steel barriers.

Response:

See revised Section 3.5.3 for the information requested.*

*See attached draft page change.

f. All openings for heating, ventilation and air conditioning system fresh air intakes (FAI) and exhausts (EXH), in buildings housing safety-related equipment, are protected against externally generated missiles by means of shield walls as indicated in Table 3.5-6. Examples are the louvred openings above the floor elevation 572'-0" in the north and south walls of the reactor building. These openings are protected by a labyrinth of missile shield walls immediately inside the opening.

3.5.3 BARRIER DESIGN PROCEDURES

The design objectives emphasize missile containment and structural integrity without secondary missile generation. 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 of scabbing. For discussion of turbine generated missiles see 3.5.1.3.

The ~~formulae~~ Ballistic Research Laboratories Formula (Reference 3.5-1) is used to determine penetration depths of missiles into steel barriers. ~~are given in 3.5.1.1.2.~~

The overall response of barriers subject to impact are investigated by the use of general energy equations given in "Introduction to Structural Dynamics", J. M. Biggs (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.

Q. 130.013
(RSP)
(3.7.1)

You state in Section 3.7.1.1 of the FSAR that a response spectrum dynamic modal analysis was performed on the reactor building structure for a Safe Shutdown Earthquake (SSE) using: (1) the design response spectra defined in Figure 3.7-1 and the damping values of Table 3.7-1 of the FSAR; and (2) the design response spectra scaled to a maximum horizontal ground acceleration of 0.25g using the damping values defined in Regulatory Guides 1.60, Revision 1, and 1.61, Revision 0, respectively. Compare the horizontal and vertical floor response spectra in the reactor building at the locations specified below using your two design criteria cited above. Additionally, provide the same comparisons for structural responses (i.e., the seismic shear, the moments, the deflections, and the floor response spectra) in the horizontal and vertical directions for all other seismic Category I structures. This comparison of the floor response spectra should be performed assuming damping values of two percent and five percent of the critical damping value, at the operating floor, the reactor stabilizer level, the reactor vessel support, the divider barrier, the basemat and the refueling hatch level for the reactor building and at the basemat, an intermediate elevation, and an upper elevation for all other seismic Category I structures.

Response:

Seismic loads used in the WNP-2 design were developed before the issuance date of Regulatory Guide 1.60, and were based on the ground response spectra as shown in Figures 3.7-1 through 3.7-4 of the FSAR. Subsequent to the issuance of Regulatory Guide 1.60, studies have been performed and are currently in process to evaluate the adequacy of the design seismic loads with respect to the structural responses obtained using Regulatory Guide 1.60 ground response spectra and realistic damping values representing soil-structure interaction effect. A detailed comparison will be provided in a future amendment.

Q. 130.014
(3.7.1)

Indicate whether and how a base line correction was made for the design time history shown in Figure 3.7-5 of the FSAR.

Response:

The design acceleration time history shown in Figure 3.7-5 is not baseline corrected. In the dynamic analyses of Seismic Category I structures, this motion is applied as ground motion at the base of the building model. The excessive displacement/velocity associated with the baseline uncorrected ground motion result in only rigid body building motion which does not induce additional accelerations, member forces and relative displacements within the structures. Consequently, the results of seismic analyses, that is, accelerations, member forces and relative displacements, are unaffected by the baseline correction.

Q. 130.015
(3.7.1)

You state in the first footnote to Table 3.7-1 of the FSAR that: "Damping values tabulated in Regulatory Guide 1.61, Revision 0, are utilized for designs which include the ground response spectra in accordance with Regulatory Guide 1.60, Revision 1, either independently or in conjunction with LOCA and S/R valve discharge related hydrodynamic loads." However, this statement contradicts your statements in Sections 3.7.1.1 and 3.7.1.3 which indicate that Regulatory Guide 1.61 was not used. Remove this ambiguity. Indicate the damping values you assumed for soil.

Response:

Please refer to the revised first footnote in Table 3.7-1 and to revised 3.7.1.3 for the information requested.*

*Draft FSAR page change attached.

damping. This ratio corresponds to a period interval varying from 0.003 seconds at a period of 0.03 seconds to 0.010 seconds at a period of 0.50 seconds. The same intervals were used in computing the response spectra at other damping values.

3.7.1.3 Critical Damping Values

The specific percentage of critical damping values used in dynamic analysis for Category I structures, systems and components are shown in Table 3.7-1 and are based on the recommendations of Reference 3.7-1. Damping values for foundation materials (soils) are also shown in Table 3.7-1.

3.7.1.4 Supporting Media for Seismic Category I Structures

The following is a description of the foundation/supporting media for Seismic Category I structures.

Structure	Average Foundation Embedment Depth (ft.)	Width of Structural Foundation (ft.)	Total Structural Height (ft.)
Reactor Building	21.5	147	265
Control Room Structure and portions of Radwaste Building*	15.5	163.5	120
Diesel-Generator Building	4	79.5	40
Standby Service Water Pumphouses	12.4	37.5	62
Spray Ponds	21	250	16

* Refer to 3.8.4.1.2 for a description of the portions of the radwaste and control building designed as Seismic Category I.

TABLE 3.7-1 Page 1 of 2

DAMPING COEFFICIENTS*

(Percent of Critical Damping)

Structure or Component**	Operating Basis Earthquake	Safe Shutdown Earthquake
Welded Steel Plate Assemblies	1.0	1.0
Welded Steel Frame Structures	2.0	2.0
Bolted or Riveted Steel Frame Structures	2.5	2.5
Reinforced Concrete Equipment Supports	2.0	3.0
Reinforced Concrete Structures	3.0	5.0
Vital Piping	0.5	1.0
Equipment	1.0	2.0

*The tabulated damping values are used in the seismic analysis in conjunction with the ground response spectra shown in Figure 3.7-1, for the design of all seismic Category I structures, systems and components.

**For structures or components combined stresses are considered below 1/2 yield for loading combinations including the operating basis earthquake, and at or near yield for loading combinations including the safe shutdown earthquake.

Q. 130.016
(3.7.2)

Indicate in Section 3.7.2.1.7 of the FSAR how the stresses due to relative displacements of supports are combined with other stresses. Indicate in 3.7.2.1.8.3, those cases where the relative displacements were insignificant and thus neglected in your analysis. Revise any other sections of the FSAR affected by your response to this item.

Response:

Please refer to revised 3.7.2.1.7 and 3.7.2.1.8.3 for the information requested.*

*Draft FSAR page changes attached.

The total inertia forces, $\underline{F}(t)$, on the structure at any time t, are obtained by adding the individual modal inertia forces at time t.

$$\underline{F(t)} = \underline{F_1(t)} + \underline{F_2(t)} + \ldots + \underline{F_n(t)} \quad \text{(Eq. 3.7.2.1-17)}$$

Once the time-histories of the displacements and inertia forces have been determined, the time-histories of internal forces, such as shears and moments, for each mode are determined by conventional structural analysis procedures. The total internal forces are obtained by adding the internal forces from each mode at each increment of time. For example, the matrix of the desired moments, $\underline{M}(t)$, is calculated from:

$$\underline{M(t)} = \underline{M_1\ (t)} + \underline{M_2\ (t)} + \ldots + \underline{M_n\ (t)} \quad \text{(Eq. 3.7.2.1-18)}$$

where M_1 (t),...,M_n (t) are the time-histories of moments in the individual modes. The maximum values of the internal forces are determined and used for design.

3.7.2.1.7 Analysis for Differential Support Displacements

Certain Seismic Category I systems (piping runs, electrical raceways and supports, duct runs, etc.), and particularly those spanning between different structures, are subject to differential support displacements. Seismic Category I system components so effected are analyzed for such effects. The relative support displacements are obtained from the dynamic analysis of structures and are imposed on the systems analyzed thus determining through a static analysis the additional stresses due to relative support displacements.

Stresses due to relative displacements of supports for piping runs are combined with other stresses as described in 3.9.3.1.1.7.

For Seismic Category I raceways and cables spanning between different structures subject to differential movements, a flexible transition is made in the system. The transition includes a slack section in cables and a flexible section in the raceways. The slack in the cable sections and flexibility in the raceway sections are sufficient to accommodate the expected differential movements.

Conduit crossing expansion joints or vibration joints in concrete slabs are provided with suitable vibration or expansion fittings to compensate for the building vibration, expansion and contraction. All runs of conduit having straight sections exceeding 100 feet in length are provided with expansion fittings so that the distance between expansion fittings does not exceed 100 feet.

For Seismic Category I ductwork spanning between different structures subject to differential movements, a flexible transition is made. The transition includes a slack section in the ductwork and sufficient flexibility in the system to accommodate the expected differential movements.

3.7.2.1.8. Dynamic Analysis of Seismic Category I Structures, Systems and Components

Seismic Category I structures, systems and components are analyzed for earthquake effects using either a response spectrum or a time-history method of dynamic modal analysis.

An alternative simplified dynamic analysis method is used for cold and/or limber piping systems, such as equipment drains and instrumentation lines. This method consists of applying constant horizontal and vertical load factors conservatively derived from floor response spectra. The load factors are derived by utilizing the response spectra for a conservatively chosen fundamental frequency based on the maximum span length of an assumed simply supported beam. These load factors are then increased to account for the multi-mode response of the piping system. Locations of pipe controls are chosen to limit the pipe span length to less than that length utilized in establishing the seismic load factors. These span lengths are also selected to limit the stresses and deflections to acceptably low values. The horizontal and vertical loading factors applied to the spans are combined in the same manner as described for the detailed dynamic analysis above.

3.7.2.1.8.3 Dynamic Analysis of Equipment

Equipment is idealized by a mathematical model consisting of lumped masses connected by elastic members or springs. Results for selected Category I equipment are given in Table 3.9-2. The dynamic response of the system is calculated by using the response spectrum method of analysis. When the equipment is supported at two or more points at different elevations, the response spectrum analysis is performed by using the response spectra at the elevation near the center of gravity of the equipment as the design spectra for the NSSS equipment, and for balance of plant using the envelope of response spectra for supports. Modal maxima are combined as described in 3.7.2.1.5. The analysis are performed assuming the horizontal ground motion to act in either of two orthogonal directions, North-South and East-West. Maximum stresses resulting from any one horizontal or vertical excitation are considered to act simultaneously and are added directly.

The relative displacements between anchors are determined from the dynamic analysis of the structures. All cases of relative displacement between anchors are considered. If significant, these relative displacements are then used in a static analysis to

determine additional stresses ~~are~~ imposed on equipment. Further details are given in 3.7.2.1.8.3.1 for the NSSS equipment and 3.7.3.9 for all other equipment. The cases where the relative displacements between anchors are insignificant and thus neglected in the analysis are those cases where the equipment is supported on a single structural element as a floor slab or wall. Typical examples where relative displacements are considered insignificant are a bank of electrical switchgear located on and anchored to a single floor slab, a diesel-generator set located on and anchored to a single isolated foundation, and an air handling unit located on and anchored to a single wall.

3.7.2.1.8.3.1 Differential Seismic Movement of Interconnected Components

The procedure for considering differential displacements for equipment anchored and supported at points with different displacement excitation is as follows:

Relative displacement between the supporting points induces additional stresses in the supported equipment. These stresses can be evaluated by performing a static analysis where each supporting point is displaced a prescribed amount. From the dynamic analysis of the complete structure, the time history of displacement at each supporting point is available. These displacements are used to calculate stresses by determining the peak nodal responses. The stresses thus obtained for each natural mode are then superimposed for all modal displacements of the structure by the square root of the sum of the squares (SRSS) method.

In the static calculation of the stresses due to relative displacements in the response spectrum method, the maximum value of the modal displacement is used. Therefore, the mathematical model of the equipment is subjected to a maximum displacement at its supporting points obtained from the modal displacements. This procedure is repeated for the significant modes (modes contributing most to the total displacement response at the supporting point) of the structure. The total stress due to relative displacement is obtained by combining the modal results using the SRSS method. Since the maximum displacement for different modes does not occur at the same time, the SRSS method is a realistic and practical method.

When a component is covered by the ASME Boiler and Pressure Vessel Code, the stresses due to relative displacement as obtained above are treated as secondary stresses.

Q. 130.017
RSP
(3.7.2)

You state in Sections 3.7.2.1.8.1 and 3.7.2.1.8.2 of the FSAR that only one horizontal component and the vertical component of seismic responses are combined in determining the maximum stresses. (Other sections of the FSAR also discuss combining only two components of a seismic response, e.g., 3.7.2.6.) It is our position that the maximum stresses be determined by combining three orthogonal sesimic components using the square-root-of-the-sum-of-the-squares (SRSS) methodology. Accordingly, provide justification for your approach. Alternatively, revise your analysis and the appropriate sections of the FSAR to reflect our position on this matter. (Refer to Section C.1.1 of Regulatory Guide 1.92, Revision 1, "Combining Modal Responses and Spatial Components in Seismic Response Analysis", February, 1976.)

Response:

As described in 3.7.2.6, the total seismic response is calculated by combining the responses from one horizontal and the vertical seismic inputs. First, for each structure, system, or component, the maximum value of stress for one horizontal component of earthquake is added to the maximum value for the vertical component of earthquake using the absolute sum method. Secondly, the maximum value for the orthogonal horizontal component of earthquake is similarly combined with the maximum value for the vertical component. The larger of the two maximums at each point in the system is used in the design. The method of combining three orthogonal seismic components using the square-root-of-the-sum-of-the-squares (SRSS) to determine the maximum stresses, as described by Regulatory Guide 1.92, Revision 1, was not a requirement for the issuance of the WNP-2 construction permit and was not utilized on WNP-2 as described in Appendix C. The combining of spatial components was performed prior to issuance of the guide and follows the method presented in the PSAR. The method used is an industry-accepted method.

Q. 130.018

You state in Section 3.7.2.1.8.2 of the FSAR that when the piping system is anchored and supported at points with different excitations, the response spectrum analysis is performed using the response spectra at or above the center of mass of the piping system for the nuclear steam supply system (NSSS) and components. However, it is our position that for systems and components supported at different locations, an envelope response spectrum be used for the analysis. Accordingly, provide justification for your approach. Alternatively, revise your analysis to reflect our position on this matter.

Response:

The analysis for NSSS piping has been revised in accordance with the NRC position. See revised Section 3.7.2.1.8.2.*

*Draft revised FSAR page attached.

3.7.2.1.8.1 Dynamic Analysis of Buildings

All Seismic Category I structures are analyzed by the response spectrum method of dynamic modal analysis and the results of the analyses are used in the design of these structures. Modal maxima are combined as described in 3.7.2.1.5. Seismic Category I structures for which floor response spectra are required are also analyzed by the time-history method of dynamic modal analysis. The analyses are performed assuming the horizontal ground motion to act in either of two orthogonal directions, North-South and East-West. Maximum stresses resulting from any one horizontal and the vertical excitations are considered to act simultaneously and are added directly.

3.7.2.1.8.2 Dynamic Analysis of Piping Systems

Each pipe line is idealized by a mathematical model consisting of lumped masses connected by elastic members. The stiffness matrix for the piping system is determined using the elastic properties of the pipe. This includes the effects of torsion, bending, shear, and axial deformations as well as change in stiffness due to curved members. The mode shapes and the undamped natural frequencies are determined. The dynamic response of the system is calculated by using the response spectrum method of analysis. ~~When the piping system is anchored and supported at points with different excitations, the response spectrum analysis is performed using the response spectra at or above the center of mass of the piping system for the NSSS systems and components. The balance of piping systems are analyzed using the envelope of response spectra for the points of attachment. Modal maxima are combined as described in 3.7.2.1.5.~~

The analyses are performed assuming the horizontal ground motion to act in either of two orthogonal directions, North-South or East-West. Maximum stresses resulting from any one horizontal or vertical excitation are considered to act simultaneously and are added directly.

The relative displacements between anchors are determined from the dynamic analysis of the structures. These relative displacements are then used in a static analysis to determine the additional stresses imposed on the piping system.

When the piping system is anchored and supported at points with different excitations the response spectrum analysis is performed using the envelope response spectrum of all attachment points. Alternatively, the multiple excitation analyses methods may be used where acceleration time histories or response spectra are applied to all piping system attachment points.

Q. 130.019
(3.7.2)

In Section 3.7.2.1.8.2 of the FSAR you state that the load factors are derived by utilizing the response spectra for a conservatively chosen fundamental frequency based on the maximum span length of an assumed simply supported beam. Provide the details of this method.

Response:

The details of the method used to determine load factors have been added to 3.7.2.1.8.2.*

*Revised FSAR draft page change attached.

3.7.2.1.8.1 Dynamic Analysis of Buildings

All Seismic Category I structures are analyzed by the response spectrum method of dynamic modal analysis and the results of the analyses are used in the design of these structures. Modal maxima are combined as described in 3.7.2.1.5. Seismic Category I structures for which floor response spectra are required are also analyzed by the time-history method of dynamic modal analysis. The analyses are performed assuming the horizontal ground motion to act in either of two orthogonal directions, North-South and East-West. Maximum stresses resulting from any one horizontal and the vertical excitations are considered to act simultaneously and are added directly.

3.7.2.1.8.2 Dynamic Analysis of Piping Systems

Each pipe line is idealized by a mathematical model consisting of lumped masses connected by elastic members. The stiffness matrix for the piping system is determined using the elastic properties of the pipe. This includes the effects of torsion, bending, shear and axial deformations as well as change in stiffness due to curved members. The mode shapes and the undamped natural frequencies are determined. The dynamic response of the system is calculated by using the response spectrum method of analysis. When the piping system is anchored and supported at points with different excitations, the response spectrum analysis is performed using the response spectra at or above the center of mass of the piping system for the NSSS systems and components. The balance of piping systems are analyzed using the envelope of response spectra for the points of attachment. Modal maxima are combined as described in 3.7.2.1.5.

The analyses are performed assuming the horizontal ground motion to act in either of two orthogonal directions, North-South or East-West. Any one of the horizontial and vertical excitations are considered to act simultaneously. Moments and forces from each of the horizontal and vertical excitations considered, are added as described in C.3., discussion for Regulatory Guide 1.92.

The relative displacements between anchors are determined from the dynamic analysis of the structures. These relative displacements are then used in a static analysis to determine the additional stresses imposed on the piping system.

An alternate simplified method of dynamic analysis is used for cold and/or limber piping systems. This is the Equivalent Static Load Method for piping. This method consists of applying constant horizontal and vertical load factors conservatively derived from seismic floor response spectra.

The description of the method is as follows: Enveloped seismic building response spectra are derived from widened seismic floor response spectra. (The widening of the building response spectra is described in 3.7.2.5).

The piping is supported seismically such that the piping fundamental frequency is higher than the building fundamental frequency, i.e., the piping system is more rigid than the building.

The piping systems are then represented by simply analytical models, e.g., simply supported beams. Initial maximum seismic support spans are analytically determined from the above model for the piping fundamental frequency. These maximum spans are modified, if required, so as not to exceed a conservative value of maximum stress based on ASME Code allowables, and a limiting piping deflection between supports.

The building accelerations at the piping frequency are determined from the enveloped seismic building response spectra. These accelerations are increased by a minimum factor of 1.15 to include the effect of higher modes of vibration. These increased accelerations (g-values) are the load factors.

The horizontal and vertical loading factors are combined in the same way as described above for the detailed dynamic analysis.

3.7.2.1.8.3 Dynamic Analysis of Equipment

Equipment is idealized by a mathematical model consisting of lumped masses connected by elastic members or springs. Results for selected Category I equipment are given in Table 3.9-2. The dynamic response of the system is calculated by using the response spectrum method of analysis. When the equipment is supported at two or more points at different elevations, the response specrum analysis is performed by using the response spectra at the elevation near the center of gravity of the equipment as the design spectra for the NSSS equipment, and for balance of plant using the envelope of response spectra for supports. Modal maxima are combined as described in 3.7.2.1.5. The analyses are performed assuming the horizontal ground motion to act in either of two orthogonal directions, North-South and East-West. Maximum stresses

resulting from any one horizontal or vertical excitation are considered to act simultaneously and are added directly.

The relative displacements between anchors are determined from the dynamic analysis of the structures. If significant, these relative displacements are then used in a static analysis to

3.7.2.1.9 Equivalent Static Load Method

This method of analysis is used for design of certain systems:

a. Systems which can be realistically represented by a single-degree-of-freedom model. The equivalent static load corresponds to the peak of the applicable (acceleration) floor response spectrum curve.

b. Systems which are structurally complex to the point they cannot be modeled realistically and yet can respond as multi-degree-of-freedom systems. The equivalent static load corresponds to 150% of the peak of the applicable (acceleration) floor response spectrum curve, to account for multi-mode response.

c. Equivalent static load method for piping is described in 3.7.2.1.8.2.

3.7.2.1.10 Dynamic Testing

When certain Seismic Category I equipment and components potential functional failure cannot be evaluated analytically (i.e., when structural integrity alone cannot assure the design intended function), dynamic testing is used to ensure operability. For example, dynamic tests of electrical items are performed in accordance with the requirements of IEEE Standard 344. (Reference 3.7-8) Test performance data and results are obtained either from previously tested comparable equipment or from the actual testing of equipment supplied. When seismic testing is impractical, a combination of test and analysis is used. Other dynamic test procedures which conservatively simulate the seismic conditions for the equipment are also used, when found acceptable by the engineer.

3.7.2.1.10.1 Equipment Testing and Test Evaluation

Category I equipment which is difficult to represent by a mathematical model or which is required to demonstrate its ability to remain operating without changing the mode of its operation (such as level switch which should not switch from "on" to "off" or vice versa during the earthquake) were subjected to actual vibration inputs on shake tables. These shake tests were performed by qualified laboratories for the equipment suppliers.

3.7.3.3.1 Field Location of Supports and Restraints

The field location of seismic supports and restraints for Seismic Category I piping and piping systems components is selected to satisfy the following two conditions:

a. The location selected must furnish the required response to control strain within allowable limits.

b. Adequate building strength for attachment of the components must be available.

The final location of seismic supports and restraints for Seismic Category I piping, piping system components, and equipment, including the placement of snubbers, is checked against the drawings and instructions issued by the engineer. An additional examination of these supports and restraining devices is made to assure that the location and characteristics of these supports and restraining devices are consistent with the dynamic and static analyses of the systems.

3.7.3.4 Basis for Selection of Frequencies

All frequencies in the 0.25 to 33 Hz range are considered in the analysis and testing of structures, systems and components. These frequencies cover the natural frequencies of most of the components and structures under consideration. If the fundamental frequency of a component is greater than or equal to 33 Hz, it is treated as rigid and analyzed accordingly. Frequencies less than 0.25 Hz are not considered, as they represent very flexible structures and are not encountered in this plant.

3.7.3.5 Use of Equivalent Static Load Method of Analysis

When the natural frequency of a structure, equipment or component is unknown, the item may be analyzed by applying a static force at the center of mass. To account for the possibility of more than one significant dynamic mode, the static force is calculated as 1.5 times the mass times the maximum acceleration from the applicable design response spectra of the point of attachments as described in 3.7.2.1.9. For structures, equipment, or components which may be realistically represented by a single degree-of-freedom system, the peak spectral accleration is used. Equivalent static load method for piping is described in 3.7.2.1.8.2.

Q. 130.020

In Section 3.7.2.1.8.3.1 of the FSAR, you state that the stresses obtained for each natural mode are superimposed for all modal displacements of the structure using the SRSS methodology. Indicate whether there are any closely-spaced modes as defined by Equation 3.7.2.1-13. If so, indicate how the responses of the closely-spaced modes were combined with other modal responses in your calculations.

Response:

The NSSS seismic design of WNP-2 was established prior to the issuance of Regulatory Guide 1.92, therefore, the consideration of closely-spaced modes in the response spectrum method of seismic analysis as described in this regulatory guide was not a requirement for the issuance of the WNP-2 construction permit. Hence, for the NSSS equipment where the response spectrum method of seismic analysis was used the closely-spaced modal responses were combined by the SRSS method. However, for all current seismic analyses of the main steam and reactor recirculation ASME Safety Class 1 piping the double sum method approved by the NRC on GESSAR 251 docket is used to combine the closely-spaced modal responses. Where the equivalent static load method and the time-history method of seismic analysis are used to calculate maximum responses it precludes the need to consider closely-spaced modes. See revised Sections 3.7.2.1.5, 3.7.2.1.8.3.1, and 3.7.2.7*.

*Draft FSAR page changes attached.

for non-NSSS piping, equipment, and structures

If several controlling frequencies in an eigenvalue solution are found to lie close together, their modal maxima are combined by direct summation (sum of absolute values method) and then combined by the square root of the sum of the squares method with other individual modal maxima. Close frequencies are considered those which satisfy the following relationship.

$$\omega_r \leq \omega_{r+1} \leq 1.10\ \omega_r \qquad \text{(Eq. 3.7.2.1-13)}$$

3.7.2.1.6 Time-History Method of Analysis

The time-history of ground acceleration, $\ddot{v}_g(t)$, is defined at discrete time intervals. The acceleration is approximated by a segmentally linear function and the solution to Duhamel's Integral (Equation 3.7.2.1-5) is obtained by using a step-by-step integration procedure (Reference 3.7-7).

$Y_r(t)$ is computed as a function of time for r = 1,2,3,..,n, where n is the number of significant modes of the system. The modal displacements, $\underline{v_r(t)}$, at the time t for the rth mode, are then calculated from:

$$\underline{v_r}(t) = \underline{\phi_r} Y_r(t) \qquad \text{(Eq. 3.7.2.1-14)}$$

The total displacements, $\underline{v(t)}$, of the structure at any time, t, are obtained by adding the individual modal displacements at time t:

$$\underline{v(t)} = \underline{v_1(t)} + \underline{v_2(t)} + \ldots + \underline{v_n(t)} \qquad \text{(Eq. 3.7.2.1-15)}$$

The inertia forces, $\underline{F_r}(t)$, at time t, for the rth mode are determined from:

$$\underline{F_r}(t) = \underline{k}\ \underline{v_r}(t) \qquad \text{(Eq. 3.7.2.1-16)}$$

For NSSS equipment and piping systems, responses for closely spaced modes were generally combined by the SRSS method, except that ~~as~~ the double-sum method, ~~as described in Regulatory Guide 1.92 and the~~ approved by the NRC on the GESSAR 251 docket, was used to combine the responses for closely spaced modes for ~~some~~ the main steam and reactor recirculation ASME Safety Class I piping. ~~seismic analyses performed after NRC approval of GESSAR 251, using response spectrum methods~~

determine additional stresses are imposed on equipment. Further details are given in 3.7.2.1.8.3.1 for the NSSS equipment and 3.7.3.9 for all other equipment.

3.7.2.1.8.3.1 Differential Seismic Movement of Interconnected Components

The procedure for considering differential displacements for equipment anchored and supported at points with different displacement excitation is as follows:

Relative displacement between the supporting points induces additional stresses in the supported equipment. These stresses can be evaluated by performing a static analysis where each supporting point is displaced a prescribed amount. From the dynamic analysis of the complete structure, the time history of displacement at each supporting point is available. These displacements are used to calculate stresses by determining the peak nodal responses. The stresses thus obtained for each natural mode are then superimposed for all modal displacements of the structure by the square root of the sum of the squares (SRSS) method including closely spaced modes, as described in 3.7.2.1.5.

In the static calculation of the stresses due to relative displacements in the response spectrum method, the maximum value of the modal displacement is used. Therefore, the mathematical model of the equipment is subjected to a maximum displacement at its supporting points obtained from the modal displacements. This procedure is repeated for the significant modes (modes contributing most to the total displacement response at the supporting point) of the structure. The total stress due to relative displacement is obtained by combining the modal results using the SRSS method. Since the maximum displacement for different modes does not occur at the same time, the SRSS method is a realistic and practical method.

When a component is covered by the ASME Boiler and Pressure Vessel Code, the stresses due to relative displacement as obtained above are treated as secondary stresses.

This method of analysis is based on the fact that the maximum resultant value of the horizontal component of the earthquake is determined when the horizontal component of the SSE is specified. This method conservatively assumes that the maximum horizontal and vertical components of the earthquake response occur simultaneously.

3.7.2.7 Combination of Modal Responses

When the response spectrum method of modal analysis is used, modal maxima are combined ~~in conformance with Regulatory Guide 1.92, Revision 1. This method is~~ described in 3.7.2.1.5. as

3.7.2.8 Interaction of Non-Category I Structures With Seismic Category I Structures

The interfaces between Category I and non-Category I structures and plant equipment have been designed for the dynamic loads and displacements of both Category I and non-Category I structures and plant equipment. In addition, all non-Category I structures meet one of the following requirements:

a. The collapse of any non-Category I structure does not cause the non-Category I structure to strike a Seismic Category I structure or component.

b. The collapse of any non-Category I structure does not impair the integrity of Seismic Category I structures or components.

c. The non-Category I structures are analyzed and designed to prevent their failure under SSE conditions in a manner such that the margin of safety of these structures is equivalent to that of Category I structures.

d. The collapse of non-Category I structures will not prevent the functioning of Seismic Category I structures or components.

Q. 130.021
(3.7.2.3)

Your response to Item 130.3 regarding the criteria used for decoupling subsystems is unclear. However, your criteria appear different from the acceptance criteria delineated in Paragraph II.3b of Section 3.7.2 of the Standard Review Plan (SRP), NUREG-75/087. Provide clarification and/or justification for your decoupling criteria.

Response:

The criteria employed for decoupling systems and subsystems in establishing the analytical models, in 3.7.2.3.1, is different from the criteria delineated in Paragraph II.3b of 3.7.2 of the SRP, NUREG-75/087.

A demonstration of the decoupling of subsystems was performed to show that the criteria of the above referenced SRP is met. The reactor pressure vessel and the primary containment vessel were modeled as part of the seismic system, and therefore were not decoupled. All major systems and components in Seismic Category I structures were checked against the criteria of the above referenced SRP and were found to meet the criteria.

Q. 130.022
(3.7.2)

It is unclear how the values of the equivalent dynamic shear modulus, G, presented in 3.7.2.4 of the FSAR were determined. Accordingly, provide the pertinent information contained in Reference 3.7-6 for all seismic Category I structures.

Response:

The pertinent information used in adopting the range of the equivalent dynamic shear modulus, G, values presented in 3.7.2.4 is contained in Section V of Reference 3.7-6. As indicated in Reference 3.7-6, the dynamic shear modulus G is computed for various depths below the building foundation by an iterative process. A strain reduced value of G is first estimated and, with the calculated average cyclic shear stress obtained by a weighting procedure based on depth and plan location, used to compute the shear strain, γ. From the computed value for γ, and the strain reduction curve (Figure 3 in Reference 3.7-6) for the appropriate depth, the strain-reduced value of G is obtained and compared with the assumed value. This process is repeated until the assumed and calculated values of G converge. An equivalent G value for the layered soil system is obtained by using the strain energy for each type of loading (vertical, sliding, rocking, torsion) as a weighting parameter. This method of evaluating compliances for layered media is based on the principle that under a given load applied to a rigid footing the contribution to the stiffness or alternatively the equivalent shear modulus of the composite system made by an individual layer is directly proportional to the strain energy contained in that layer, and on the assumption that the strain energy in an individual layer is approximately equal to that in a corresponding region of a homogeneous elastic half-space having a shear modulus and Poisson's ratio equal to that layer. The assumption is founded on the premise that the stress distribution in a layered medium is approximately the same as that in a homogeneous elastic half-space. Therefore, if it is

assumed that the layered and homogeneous media have identical Poisson's ratios and also incur identical stress fields, then the strain energy in the layered media may be found by scaling the strain energy in the homogeneous solid in proportion to the reciprocal of the shear modulus of each layer.

Q. 130.023
(3.7.2)

Identify in Section 3.7.2.5 of the FSAR, those cases where the peaks of floor response spectra were widened by less than 15 percent of the structural frequencies. (Refer to Paragraph II.9 of Section 3.7.2 of the Standard Review Plan.)

Response:

Please refer to revised 3.7.2.5 for the information requested.*

*Draft FSAR page change attached.

The floor response spectra are computed for the safe shutdown earthquake and the operating basis earthquake for the two horizontal orthogonal directions and the vertical direction.

Spectral values, (the maximum response of a single degree-of-freedom oscillator) are obtained using a step-by-step integration (Reference 3.7-7). The analytical solution assumes that the acceleration histories of structural response are linear within the time interval of 0.02 seconds. The intergration is performed at either the 0.02 second time intervals or at 0.05 times the natural period of the single degree-of-freedom oscillator, whichever is smaller.

The discrete periods or frequencies used in the calculation of the floor response spectra are in compliance with the values suggested in Regulatory Guide 1.122, September 1976.

To account for variations in structural frequencies, the peaks of the computed floor response spectra associated with each of the structural frequencies are widened by no less than ± 15% for all seismic Category I structures, except the primary metal containment, the reactor pressure vessel (RPV), the RPV pedestal and the sacrificial shield wall (SSW). For the primary metal containment vessel, the RPV, the RPV pedestal and the SSW, the reponse spectra is widened by no less than ± 10%.

~~To account for variations in structural frequencies, the peaks of the computed floor response spectra associated with each of the structural frequencies are widened by ± 15% in most cases and never less than ±10%.~~

3.7.2.6 Three Components of Earthquake Motion

The use of three components of earthquake motion, as described by Regulatory Guide 1.92, Revision 1, was not a requirement for the issuance of the WNP-2 construction permit. The total seismic response is calculated by combining the response calculated from analyses due to one horizontal and one vertical seismic input.

Two sets of seismic results are obtained. First the maximum value of the horizontal component of the earthquake is assumed to act in one horizontal direction simultaneously with the vertical component and the loads are computed for this combination.

The maximum value of the horizontal component of the earthquake is assumed to act perpendicular to the direction previously assumed and simultaneously with the vertical component and loads are computed for this combination. The larger of these two loads, at each point in the system, is used for design.

Q. 130.024
(3.7.2)

For non-symmetric structures, there may be dynamic coupling between translational and torsional motions. As a result, calculating the torsional moment as the product of the inertial force and the distance between the centers of mass and rigidity may or may not be adequate. Accordingly, provide in Section 3.7.2.11 of the FSAR, the bases for your approach. Indicate how the torsional effects were included in the generation of the floor response spectra.

Response:

The present torsional moments are calculated according to the criteria in 3.7.2.11 as the product of the inertial force and the distance between the center of mass and the center of rigidity.

Calculation of torsional effects using a dynamic analysis that considers coupled translational and torsional degrees of freedom as referred to in the question is currently underway, to compare the new torsional effects with those of the present torsional moments and for verification of structural adequacy. The results of this reanalysis will be reported in a future amendment.

Q. 130.025
(3.7.2)

In Section 3.7.2.12 of the FSAR, you state that Table 3.7-17 contains comparisons of the seismic responses obtained from the response spectrum approach and the time history approach. However, this comparison is not provided. Accordingly, provide this information.

Response:

The statement in 3.7.2.12 that Table 3.7-17 contains comparisons of the calculated load (response) in the reactor pressure vessel and internals was inadvertently misplaced in 3.7.2.12 and is now deleted. This statement also appears in 3.7.3.14 and has been revised for further clarification.*

*See attached draft FSAR page changes.

inertial force applied at the center of mass of each floor above and a lever arm equal to the distance from the center of mass of the floor to the center of rigidity of the story. The lever arm is not less than the minimum eccentricity required by the Uniform Building Code. (Reference 3.7-9) The torsional moment and story shear are distributed to the resisting elements in accordance with the provisions of the Uniform Building Code.

Symmetrical structures are analyzed in a similar manner for torsional effects using minimum eccentricities between the center of mass and the center of rigidity as defined by the Uniform Building Code.

3.7.2.12 Comparison of Responses

Comparisons of structural responses (accelerations) of the reactor building obtained using: (a) the response spectrum method with the site design response spectra and, (b) the time history method with the simulated time-history of earthquake acceleration described in 3.7.1.2 are presented in Figure 3.7-25. These results illustrate the conservatism inherent in the simulation process illustrated by Figure 3.7-6 through 3.7-10, and carried over in calculation of floor response spectra used in seismic design of systems, components and equipment. A more appropriate comparison is obtained between structural responses (acceleration) of the reactor building obtained using: (a) the response spectrum method with response spectra calculated from the simulated earthquake acceleration, and (b) the time-history method. also These are also displayed in Figure 3.7-25 and show good agreement between the results obtained using the two methods of dynamic modal analysis.

3.7.2.13 Methods for Seismic Analysis of Dams

No Seismic Category I dams are utilized in this facility.

3.7.2.14 Determination of Seismic Category I Structure Overturning Moments

Overturning and sliding effects of horizontal seismic loadings have been considered in combination with the effects of vertical seismic loadings. The seismic loads used consist of two horizontal orthogonal and vertical components of earthquake motions. Each horizontal component is taken separately and is applied concurrently with the vertical component. For Seismic Category I structures, the results of the dynamic analysis have been converted to equivalent static loads at the mass points.

3.7.3.14 Seismic Analyses for Reactor Internals

The seismic analysis of the reactor is described in 3.7.2.3.4. A comparison of the maximum calculated seismic loads and the allowable seismic loads in the reactor pressure vessel and internals is given in Table 3.7-17. The damping values are given in Table 3.7-1.

3.7.3.15 Analysis Procedure for Damping

Damping values used for Category I subsystems are discussed in 3.7.2.15.

3.7.4 SEISMIC INSTRUMENTATION

3.7.4.1 Comparison With Regulatory Guide 1.12

The seismic instrumentation system for WNP-2 complies with the requirements of USNRC Regulatory Guide 1.12, Revision 1, as described below.

3.7.4.2 Location and Description of Instrumentation

Triaxial strong-motion accelerographs are installed at appropriate locations to provide data on the seismic input to containment, data on the frequency, amplitude and phase relationship of the seismic response of the containment structure and to provide data on the seismic input to other Category I structures, systems and components. The criteria for selection of Category I structures, components, and equipment to be instrumented, and the location of instrumentation, is that which will enable the evaluation of the following:

a. To determine if the input design response spectra has been exceeded.

b. To determine if the calculated resultant vibratory responses used in the design of the representative Category I structures and equipment have been exceeded.

c. The degree of applicability of the mathematical models used in the seismic analysis of the buildings and equipment.

Q. 130.026
(3.7.2)
(130.006)

Your response to Question 130.006 is not satisfactory. Accordingly, provide in Section 3.7.2.14 of the FSAR the details of your procedures to calculate the capability of safety-related structures to resist overturning and sliding. Additionally, provide the factors of safety against overturning and sliding for each safety-related structure due to horizontal loads.

Response:

Refer to revised 3.7.2.14 for the information requested.*

See the response to Question 130.044 for the factors of safety.

*Draft FSAR page changes attached.

inertial force applied at the center of mass of each floor above and a lever arm equal to the distance from the center of mass of the floor to the center of rigidity of the story. The lever arm is not less than the minimum eccentricity required by the Uniform Building Code. (Reference 3.7-9) The torsional moment and story shear are distributed to the resisting elements in accordance with the provisions of the Uniform Building Code.

Symmetrical structures are analyzed in a similar manner for torsional effects using minimum eccentricities between the center of mass and the center of rigidity as defined by the Uniform Building Code.

3.7.2.12 Comparison of Responses

Comparisons of structural responses (accelerations) of the reactor building obtained using: (a) the response spectrum method with the site design response spectra and, (b) the time history method with the simulated time-history of earthquake acceleration described in 3.7.1.2 are presented in Figure 3.7-25. These results illustrate the conservatism inherent in the simulation process illustrated by Figure 3.7-6 through 3.7-10, and carried over in calculation of floor response spectra used in seismic design of systems, components and equipment. A more appropriate comparison is obtained between structural responses (acceleration) of the reactor building obtained using: (a) the response spectrum method with response spectra calculated from the simulated earthquake acceleration, and (b) the time-history method. These are displayed in Figure 3.7-25 and show good agreement between the results obtained using the two methods of dynamic modal analysis.

3.7.2.13 Methods for Seismic Analysis of Dams

No Seismic Category I dams are utilized in this facility.

3.7.2.14 Determination of Seismic Category I Structure Overturning Moments

Overturning and sliding effects of horizontal seismic loadings have been considered in combination with the effects of vertical seismic loadings. The seismic loads used consist of two horizontal orthogonal and vertical components of earthquake motions. Each horizontal component is taken separately and is applied concurrently with the vertical component. For Seismic Category I structures, the results of the dynamic analysis have been converted to equivalent static loads at the mass points.

Seismic Category I structures are located above the present groundwater elevation of 380 feet MSL. However, uplift and lateral hydrostatic pressures are considered, taking into account the maximum groundwater elevation of 420 feet MSL in the event the Ben Franklin Dam is constructed, as discussed in 3.4. The uplift and hydrostatic pressures, including seismic effects due to dry and saturated soils, as applicable, are applied concurrently.

To calculate the capability of safety-related structures to resist overturning, the following load combinations are considered:

a. D + E + Q* + Uplift

b. D + W + Q + Uplift

c. D + E^1 + Q* + Uplift

d. D + W^1 + Q + Uplift

Load combination c. above is used since the resulting horizontal and vertical forces produce the maximum overturning effects. The load terms in the load combinations are defined in 3.8.4.3.3, except as follows:

a. The dead load, D, also includes the weight of dry and saturated backfill, as applicable.

b. The uplift force, not included in 3.8.4.3.3, is taken as the weight of the water displaced by the structure, acting vertically upward and applied to the bottom surface of the basemat.

The overturning moments and the stabilizing moments are calculated about the lower edge (toe) of the basement. The safety factor against overturning is calculated by dividing the total stabilizing moment by the total overturning moment.

To calculate the capability of safety-related structures to resist sliding, the load combinations considered are the same as those listed above for calculating the capability of safety-related structures to resist overturning. Load combination c. produces the maximum sliding effects and is used. The safety-factor against sliding is calculated by dividing the frictional force resisting sliding between the basemat and the soil by the summation of horizontal forces causing sliding.

The factors of safety against overturning and sliding for safety-related structures are tabulated in 3.8.5.5.

3.7.2.15 Analysis Procedure for Damping

For structures and components, damping coefficients are selected as discussed in 3.7.1.3.

For the foundation materials, either internal or radiation damping or both are considered. The horizontal translation, vertical translation, and rocking motion damping values are determined as described in Reference 3.7-5.

The selected design values are significantly smaller than the calculated values (see Table 3.7-2).

For composite structures made up from different materials, when the various components cannot be decoupled due to interaction effects, an approximate weighted average damping value is used for each mode of vibration of the structure. This is accomplished by breaking the mode shapes into their various components, then assigning a damping value to each component depending upon the principal action of this component. A weighted average value is then determined for the particular mode under consideration. In this manner, a composite damping value is determined for each mode and the total response is calculated in the regular manner.

For example, for horizontal motions the weighted average damping value for the n^{th} mode will be obtained from the following relation:

$$D_n = \frac{D_s E_{sn} + D_h E_{hn} + D_r E_{rn}}{E_{sn} + E_{hn} + E_{rn}} \qquad \text{(Eq. 3.7.2.15-1)}$$

where:

D_n = Weighted average damping for the n^{th} mode

D_s = Damping ratio for the superstructure

D_h = Damping ratio for the horizontal translation

D_r = Damping ratio for the rocking motion

Q. 130.027
(3.7.2)

In Section 3.7.2.15 of the FSAR, you state that the design values of the damping coefficients in Table 3.7-2 are significantly smaller than the calculated values. However, this information is not in this table. Accordingly, provide this information and your bases for Equation 3.7.2.15-1.

Response:

Damping coefficients for foundation materials are limited to the design values given in Table 3.7-1. However, realistic damping values calculated on the basis of the elastic half space theory are significantly higher than the design values used. Please see revised Section 3.7.2.15 for the information requested.*

The calculated damping coefficients are given in the following tabulation for the listed Seismic Category I buildings:

ELASTIC HALF SPACE DAMPING RATIO
(PERCENT OF CRITICAL DAMPING)

Building	Vertical	Horizontal	Rocking
Reactor Building	66.5	32.0	7.6
Radwaste and Control Building	93.5	59.5	65.0
Diesel-Generator Building	63.0	38.0	22.0

Please refer to revised 3.7.2.15 for the basis of Equation 3.7.2.15-1.*

*Draft FSAR page changes attached.

Seismic Category I structures are located above the present groundwater elevation of 380 feet MSL. However, uplift and lateral hydrostatic pressures are considered, taking into account the maximum groundwater elevation of 420 feet MSL in the event the Ben Franklin Dam is constructed, as discussed in 3.4. The uplift and hydrostatic pressures, including seismic effects due to dry and saturated soils, as applicable, are applied concurrently.

Conservative calculations of overturning and base shears have been made in checking the general stability of Seismic Category I structures using load combinations involving the seismic loads and the uplift and lateral hydrostatic pressures as discussed above.

3.7.2.15 Analysis Procedure for Damping

For structures and components, damping coefficients are selected as discussed in 3.7.1.3.

For the foundation materials, either internal or radiation damping or both are considered. The horizontal translation, vertical translation, and rocking motion damping values are determined as described in Reference 3.7-5.

The selected design values are significantly smaller than the calculated values. Table 3.7-1 presents the design values of the damping coefficients used. The formulae presented in Table 3.7-2 can be used to calculate the realistic damping coefficients.

For composite structures made up from different materials, when the various components cannot be decoupled due to interaction effects, an approximate weighted average damping value is used for each mode of vibration of the structure. This is accomplished by breaking the mode shapes into their various components, then assigning a damping value to each component depending upon the principal action of this component. A weighted average value is then determined for the particular mode under consideration. In this manner, a composite damping value is determined for each mode and the total response is calculated in the regular manner.

For example, for horizontal motions the weighted average damping value for the n^{th} mode will be obtained from the following relation:

$$D_n = \frac{D_s E_{sn} + D_h E_{hn} + D_r E_{rn}}{E_{sn} + E_{hn} + E_{rn}} \qquad \text{(Eq. 3.7.2.15-1)}$$

where:

D_n = Weighted average damping for the n^{th} mode

D_s = Damping ratio for the superstructure

D_h = Damping ratio for the horizontal translation

D_r = Damping ratio for the rocking motion

E_{sn} = Energy stored in the superstructure

E_{hn} = Energy stored in the horizontal spring

E_{rn} = Energy stored in the rocking spring

The basis for Equation 3.7.2.15-1 is presented as Equation (4) in Reference 3.7-15.

In a linear dynamic analysis for the NSSS Systems and components the procedure to be utilized to properly account for damping in different elements of a coupled system model is as follows:

a. A structural damping of the various structural elements of the model are first specified. Each value is referred to as the damping ratio (B_j) of a particular component which contributes to the complete stiffness of the system.

b. Perform a modal analysis of the linear system model. This will result in a modal matrix (ϕ) normalized such that $\phi_i^T K\phi_i = W_i^2$, where K is the stiffness matrix, W_i the circular natural frequency of mode i and ϕ_i^T is the transpose ϕ, which is a column vector of ϕ corresponding to the mode shape of mode i. Matrix ϕ contains all translational and rotational coordinates.

c. Using the strain energy of the individual components as a weighting function, the following equation can be derived to obtain a suitable damping ratio (B_i) for the i^{th} mode.

$$B_i = \frac{\sum_{j=i}^{N} \phi_{ij}^{T} B_j K_j \phi_{ij}}{w_i^2} \qquad \text{(Eq. 3.7.2.15-2)}$$

where:

N = Total number of structural elements

ϕ_{ij} = Mode shape for mode i (ϕ_i^T as transpose)

B_j = Percent damping associated with element j

3.7-8 Institute of Electrical and Electronic Engineers (IEEE) Standard 344, "Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations".

3.7-9 ICBO, "Uniform Building Code" International Conference of Building Officials, Whittier, Ca., (1970)

3.7-10 "BWR/6 General Electric Standard Safety Analysis Report" (Gessar), Vol. 1, General Electric Co., San Jose, Ca., 4/30/74.

3.7-11 Liu, L.K., Seismic Analysis of the Boiling Water Reactor, Symposium or Seismic Analysis of Pressure Vessel and Piping Components, First National Congress on Pressure Vessel and Piping, San Francisco, Ca., May 1971.

3.7-12 Newmark, N.M., Earhtquake Response Analysis of Reactor Structures, Paper Presented Before 1st International Conference on Structural Mechanics, Berlin, September, 1971.

3.7-13 Shah, H.H., and Chu, S.L., "Seismic Analysis of Underground Structural Elements", Journal of the Power Division, ASCE, No. P01 (July, 1974).

3.7-14 Iqbal, A., and Goodling, E.C., Seismic Design of Buried Piping, Paper Presented Before ASCE Speciality Conference on Structural Design of Nuclear Plant Facilities, New Orleans, Louisiana, December, 1975.

3.7-15 Whitman, R.V., "Soil-Structure Interaction", Seismic Design of Nuclear Power Plants, R.J. Hansen, editor, The M.I.T. Press, Cambridge, Massachusetts, and London, England, 1970, pp. 245-269.

Q. 130.028
(3.7.3)

It is not apparent to us which of your references is the Reference 10 that you cite in Section 3.7.3.2.1 of the FSAR. Accordingly, provide justification for the number of earthquake cycles you estimate in this section.

Response:

The Reference 10 cited in 3.7.3.2.1 refers to Reference 3.7-10. Section 3.7.3.2.1 has been appropriately revised.*

*See attached draft FSAR pages.

K_j = Stiffness contribution of element j

W_i = Circular natural frequency of mode i

3.7.3 SEISMIC SUBSYSTEM ANALYSIS

The general approach to the seismic subsystem analysis is identical to those procedures described in 3.7.2 for seismic system analysis, except for the soil/structure interaction effects.

3.7.3.1 Seismic Analysis Methods

The seismic analysis method used to analyze Seismic Category I subsystems is described in 3.7.2.1.

3.7.3.2 Determination of Number of Earthquake Cycles

3.7.3.2.1 Number of Cycles for All Items Except NSSS Systems and Components

Assuming the mathematical model of strong motion earthquake acceleration described in 3.7.1.2 (T=15 seconds), the number of peaks and troughs, N, of the random process representing the structural response may be estimated (Reference 3.7-3). The response of nuclear plant structures is controlled mostly by one governing mode, for the range of building frequencies normally encountered in nuclear plant facilities, (1.5 Hz to 6.0 Hz,) the number N is evaluated to be from 50 to 150. For a strong motion earthquake acceleration of 30 seconds in duration, N is from 100 to 300 for each seismic event. Fatigue evaluation due to a safe shutdown earthquake is not required by ASME Code, Section III since it qualifies as a faulted condition.

The operating basis earthquake is an upset condition and therefore must be included in fatigue evaluations according to ASME Code, Section III. The probability for the occurrence of a seismic event of OBE intensity is extremely low. Lower intensity earthquakes have a higher probability of occurrence. Based on Reference 3.7-10, which summarizes data related to seismic histories presented in PSARs for many plants, it is conservatively assumed that combined effects due to seismic events of an intensity less than or equal to OBE intensity may be considered equivalent to two earthquakes of OBE

intensity. Therefore, the lifetime number of earthquake cycles may range from 200 to 600 assuming 30 seconds of strong motion earthquake acceleration for each seismic event.

During an actual seismic disturbance, only a small percentage of these cycles occur at the maximum, or even at a significant stress level. Reference 3.7-10 states that 99.5% of the stress reversals occur below 75% of the maximum stress level, and 95% of the reversals lie below 50% of the maximum stress level (See Figure 3.7-26). Based on this data, it is assumed that a total lifetime number of maximum seismic load cycles of 60 is a conservative estimate of the number of cycles which will have a significant contribution to fatigue usage.

Delete

~~During an actual seismic disturbance, only a small percentage of these cycles occur at the maximum, or even at a significant stress level. Reference 10 states that 99.5% of the stress reversals occur below 75% of the maximum stress level, and 95% of the reversals lie below 50% of the maximum stress level (See Figure 3.7-26). Based on this data, it is assumed that a total lifetime number of maximum seismic load cycles of 60 is a conservative estimate of the number of cycles which will have a significant contribution to fatigue usage.~~

3.7.3.2.2 Number of Cycles for NSSS Systems and Components

To evaluate the number of cycles which exist within a given earthquake, a typical boiling water reactor building-reactor dynamic model was excited by three different recorded time histories: (a) May 18, 1940, El Centro NS component 29.4 sec; (b) 1952, Taft N 69° W component, 30 sec; and (c) March 1957, Golden Gate S80E component, 13.2 sec. The modal response was truncated such that the response of three different frequency bandwidths could be studied, (0+-10 Hz, 10-20 Hz, and 20-50 Hz). This was done to provide a good approximation to the cyclic behavior expected from structures with different frequency content.

Enveloping the results from the three earthquakes and averaging the results from several different points of the dynamic model, the cyclic behavior, as given in Table 3.7-18, was formed.

Independent of earthquake or component frequency, 99.5% of the stress reversals occur below 75% of the maximum stress level, and 95% of the reversals lie below 50% of the maximum stress level. This relationship is graphically shown in Figure 3.7-26.

Q. 130.029
(3.7.3)

You present two conditions in Section 3.7.3.3 of the FSAR which must be satisfied in selecting the field locations of seismic supports and restraints. Elaborate on how these two objectives were met.

Response:

Please refer to revised 3.7.3.3.1 for the information requested.*

*Draft FSAR page changes attached.

3.7.3.3.1 Field Location of Supports and Restraints

The field location of seismic supports and restraints for Seismic Category I piping and piping systems components is selected to satisfy the following two conditions:

a. Restraint locations are chosen sufficiently close to each other to limit the stress and strain of the piping system to acceptable values. Spring supports are neglected for seismic analysis. Seismic restraints are constructed sufficiently rigid so as to preclude interaction with the piping system.

b. Structures are provided of sufficient capacity to support the seismic supports and restraints and to withstand the seismic and/or LOCA loads transferred to the supporting structure by the seismic support and/or restraint. The applicable load combinations in Tables 3.8-10, 3.8-11, 3.8-15 and 3.8-16 are used, depending on the loading conditions to which the structure could be subjected.

The final location of seismic supports and restraints for Seismic Category I piping, piping system components, and equipment, including the placement of snubbers, is checked against the drawings and instructions issued by the engineer. An additional examination of these supports and restraining devices is made to assure that the location and characteristics of these supports and restraining devices are consistent with the dynamic and static analyses of the systems.

3.7.3.4 Basis for Selection of Frequencies

All frequencies in the 0.25 to 33 Hz range are considered in the analysis and testing of structures, systems and components. These frequencies cover the natural frequencies of most of the components and structures under consideration. If the fundamental frequency of a component is greater than or equal to 33 Hz, it is treated as rigid and analyzed accordingly. Frequencies less than 0.25 Hz are not considered, as they represent very flexible structures and are not encountered in this plant.

3.7.3.5 Use of Equivalent Static Load Method of Analysis

When the natural frequency of a structure, equipment or component is unknown, the item may be analyzed by applying a static force at the center of mass. To account for the possibility of more than one significant dynamic mode, the

static force is calculated as 1.5 times the mass times the maximum acceleration from the applicable design response spectra of the point of attachments as described in 3.7.2.1.9. For structures, equipment, or components which may be realistically represented by a single degree-of-freedom system, the peak spectral acceleration is used.

Q. 130.030
(3.7.3)

The methods of analysis described in Sections 3.7.3.12.1.3 and 3.7.3.12.1.5 of the FSAR are unacceptable. The procedure for predicting the seismic stresses in buried pipes suggested by Newmark (Reference 3.7-12) is based on the assumption that pipes will move with the soil during an earthquake; i.e., there will be no slippage at the soil-pipe interface. You have not provided a basis for your proposed extension of these procedures to include consideration of friction and slippage (e.g., Equations 3.7.3.12.1.3-2 through 3.7.3.12.1.3-5 and Equation 3.7.3.12.1.5-3). Accordingly, it is our position that you use the procedures in Reference 3.7-12. Alternatively, provide either theoretical or experimental justifications for the equations cited above. Identify the values of "Vm" used in your analysis and explain how they were determined.

Response:

The procedure suggested by N. M. Newmark in Reference 3.7-12 is used for predicting seismic stresses in buried pipes. Please refer to revised 3.7.3.12.1.3 and 3.7.3.12.1.4 for the information requested. Other portions revised to reflect the changes in 3.7.3.12.1.3 and 3.7.3.12.1.4 are: 3.7.3.12, 3.7.3.12.1.1, 3.7.3.12.1.2, 3.7.3.12.1.5, 3.7.5 and Figures 3.7-27 and 3.7-28.*

*Draft revised FSAR page changes attached.

The stress criteria are defined in 3.8, 3.9 and 3.10.

3.7.3.10 Use of Constant Vertical Static Factors

The use of constant vertical static factors, as applied to Seismic Category I subsystems, is limited, as discussed in 3.7.2.10.

3.7.3.11 Torsional Effects of Eccentric Masses

Torsional effects of eccentric masses are discussed in 3.7.2.3.2.

When the torsional effect of an eccentric mass is likely to have a significant effect on the result of an analysis, the eccentric mass is included in the analytical mode. If the pipe stresses due to an eccentric mass are expected to be insignificant, the offset moment due to the eccentric mass is usually neglected.

3.7.3.12 Buried Seismic Category I Piping Systems and Tunnels

Seismic Category I piping penetrating exterior building foundation walls are furnished with oversized wall sleeves and flexible closure boots as shown in Figures 3.8-48 and 3.8-49.

No buried Seismic Category I tunnels are utilized in this facility.

Underground nuclear safety related piping is designed to safely resist operating loads and loads due to accident conditions which include seismic waves passing through the soil media supporting these elements and relative seismic displacements between building and surrounding soil. Analysis of these underground pipes subjected to ground motion is based on their boundary conditions and the elastic properties of the soil and piping.

3.7.3.12.1 Procedures for Predicting the Stresses of Buried Pipes in the Free Field

3.7.3.12.1.1 Method of Analysis

The method of analysis developed is based on Reference 3.7-12.

3.7.3.12.1.2 DELETED

3.7.3.12.1.3 Axial Stresses in Pipe

The method of analysis as suggested by N. M. Newmark in Reference 3.7-12 is used in the analysis of the axial stresses in buried pipes. The value of the maximum particle velocity, V_m, used in the analysis is calculated by following the recommendations of N. M. Newmark, et al, in Reference 3.7-2.

3.7.3.12.1.4 Bending Stresses in Pipes

The method of analysis as suggested by N. M. Newmark in Reference 3.7-12 is used in the analysis of the bending stresses in buried pipes.

3.7.3.12.1.5 DELETED

(DELETED)

3.7.3.12.1.6 Buried Piping Encased in Oversized Culvert Sections

Certain portions of buried pipes are encased in oversized culverts. The encasement serves the dual purpose of providing protection against damage of piping under heavily loaded areas such as roads, and of accommodating thermal expansion at changes in direction of the piping.

The encased piping does not come in contact with the soil and can thus be analyzed by the same methods used for piping in free space. The dead load, internal pressure, seismic and thermal stresses are maintained below allowable limits.

3.7-8 Institute of Electrical and Electronic Engineers (IEEE) Standard 344, "Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations".

3.7-9 ICBO, "Uniform Building Code" International Conference of Building Officials, Whittier, Ca., (1970)

3.7-10 "BWR/6 General Electric Standard Safety Analysis Report" (Gessar), Vol. 1, General Electric Co., San Jose, Ca., 4/30/74.

3.7-11 Liu, L.K., Seismic Analysis of the Boiling Water Reactor, Symposium or Seismic Analysis of Pressure Vessel and Piping Components, First National Congress on Pressure Vessel and Piping, San Francisco, Ca., May 1971.

3.7-12 Newmark, N.M., Earthquake Response Analysis of Reactor Structures, Paper Presented Before 1st International Conference on Structural Mechanics, Berlin, September, 1971.

3.7-13 (DELETED)

3.7-14 (DELETED)

DELETED

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2		FIGURE 3.7-27

DELETED

WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2		FIGURE 3.7-28

TABLE OF CONTENTS (Continued)

LIST OF FIGURES (Continued)

Q. 130.031
(3.7.3.12)

Indicate the clearance between the pipes and sleeves shown in Figures 3.8-48 and 3.8-49 of the FSAR and indicate the differential displacements of these buried pipes relative to the buildings. Provide the details of your procedures for estimating the stresses in those portions of the buried pipes connected to the buildings.

Response:

Please refer to revised 3.7.3.12 for the information requested.*

*Revised FSAR page changes attached.

The stress criteria are defined in 3.8, 3.9 and 3.10.

3.7.3.10 Use of Constant Vertical Static Factors

The use of constant vertical static factors, as applied to Seismic Category I subsystems, is limited, as discussed in 3.7.2.10.

3.7.3.11 Torsional Effects of Eccentric Masses

Torsional effects of eccentric masses are discussed in 3.7.2.3.2.

When the torsional effect of an eccentric mass is likely to have a significant effect on the result of an analysis, the eccentric mass is included in the analytical mode. If the pipe stresses due to an eccentric mass are expected to be insignificant, the offset moment due to the eccentric mass is usually neglected.

3.7.3.12 Buried Seismic Category I Piping Systems and Tunnels

Seismic Category I piping penetrating exterior building foundation walls are furnished with oversized wall sleeves and flexible closure boots as shown in Figures 3.8-48 and 3.8-49. The clearance between the pipes and wall sleeves in Figures 3.8-48 and 3.8-49 is 1.6 inches, minimum, for all penetrations except for three penetrations where the clearance is 1.25 inches. The differential displacements of these buried pipes relative to the buildings are calculated to be less than 1.22 inches. These differential displacements include the effects of both the SSE and the building settlements.

No buried Seismic Category I tunnels are utilized in this facility.

Underground nuclear safety related piping is designed to safely resist operating loads and loads due to accident conditions which include seismic waves passing through the soil media supporting these elements and relative seismic displacements between building and surrounding soil. Analysis of these underground pipes subjected to ground motion is based on their configuration and boundary conditions and the elastic properties of the soil and piping.

For the stress analysis of the portions of the buried pipes penetrating the wall sleeves and connected to the buildings, the relative displacement between the building and soil is imposed on the buried pipe in addition to the pipe internal pressure, pipe dead weight and seismic and thermal effects on

the pipe. When the piping is enclosed in encapsulated sleeves, as shown in Figure 3.8-49, the supports inside the encapsulated sleeves are also modeled in the analysis of the piping system.

3.7.3.12.1 Procedures for Predicting the Stresses of Buried Pipes in the Free Field

3.7.3.12.1.1 Method of Analysis

The method of analysis developed is based on References 3.7-12, 3.7-13 and 3.7-14.

Q. 130.032
(3.8.2)

Provide in Section 3.8.2.1 of the FSAR, the mechanical properties of the two-inch thick, compressible, isolation material between the steel containment vessel and the reinforced concrete biological shield. Indicate if the properties of this compressible material could be affected by radiation.

Response:

See revised 3.8.2.1 (page 3.8-9).*

*Draft FSAR page change attached.

steel shell temperature of 340°F, and an internal pressure of 45 psig following a postulated reactor loss-of-coolant accident.

g. The above design, materials, and construction of the containment vessel expansion gap provides sufficient space for thermal expansion of the steel containment vessel shell. Moreover, this method of construction prevents either concrete or other foreign material from entering and/or reducing the gap. Local stress areas are thereby prevented, and the primary containment system is capable of accommodating both normal operating as well as postulated accident conditions.

The steel primary containment vessel, including all penetrations and welded attachments, is designed to act as a structural component within the reactor building as described in 3.8.2.4. The general configuration, datum and elevations are shown in Figures 3.8-1 and 3.8-2. The primary containment vessel is provided with two concentric circular skirts on the bottom ellipsoidal head integral with the vessel. The skirts are anchor bolted to the concrete foundation mat. The bottom ellipsoidal head and the upper portion of the skirts connected to the head are considered part of containment pressure boundary in accordance with the ASME Code, Section III. The lower portion of the skirts follow the requirements of the AISC Code. The skirts are backed up by concrete fill. The concrete fill and the concrete foundation mat, discussed in 3.8.5, are not part of the containment vessel.

The external compressive stress applied to the containment vessel by the polyurethane foam, due to vessel thermal expansion from ambient conditions, ranges from 1.2 to 1.5 psi at normal operating and LOCA temperatures. At these temperatures the stress-strain curve for the polyurethane foam is nearly linear between 5% and 60% compression, ranging from approximately 1.2 psi at 5% compression to 1.8 psi at 60% compression. These properties are not significantly affected by the level of radiation exposure received over the plant life.

Q. 130.033
(3.8.2)

You state in 3.8.2.1 of the FSAR that the jet impingement force of 534 kips discussed in 3.8.2.3.5.2 might cause local yielding of the drywell shell but that a plastic analysis in accordance with Section III of the ASME Code demonstrates that rupture of the drywell shell will not occur. You conclude on this basis that the jet impingement force will not adversely affect the leakage rate from the primary containment. Provide the details of the cited analysis and indicate the value of the calculated ductility ratio. Provide the basis for your conclusion that the leak tightness of the containment vessel will not be adversely affected.

Response:

Please refer to revised 3.8.2.1.*

*Draft FSAR page changes attached.

In accordance with the requirements of the ASME Code, Section III, for the normal operating condition, (Normal Condition) a fatigue analysis was performed.

Under normal operating condition (Normal Condition) a fatigue analysis is performed in accordance with the requirements of ASME Code, Section III.

Under emergency condition, the jet impingment force of 534 kips as outlined in 3.8.2.3 might cause local yielding of the drywell shell. An analysis (plastic analysis in accordance with the requirements of the ASME Code, Section III) demonstrates that rupture will not occur. Local deformation caused by the jet impingment force does not affect the leak tightness of the containment vessel.

The analysis of the jet impingement effects on the primary containment vessel (Ref. 3.8-21) is briefly summarized as follows:

1. Phase 1 - The conical region of the containment shell was modeled and a general shell of revolution analysis was performed using the HYBOS Computer Program (Ref. 3.8-22).

   Two critical locations were chosen for independent application of the jet force. One, located approximately in the middle of two box ring stiffeners is a logical candidate for maximum deflection; the second, located on the thinnest section nearly adjacent to a stiffener, is a location where largest curvatures could occur if the shell contacts the concrete biological shield wall spaced two inches from the shell.

   Since the impinged area (429 square inches) subtends only a small arc of the total periphery, a Fourier harmonic expansion of 11 terms is used to represent the jet force of 534 kips.

   The response to gravity, static seismic and design pressure loads were also computed. The results of the most severe combinations of the loads (Appendix C and D of Ref. 3.8-21) show that the shell will contact the concrete for either candidate jet force location; consequently the elastic analysis is not valid in the immediate area of the jet load. The largest computed stresses were found for the second location and exceeded yield; therefore, an elastic-plastic analysis was next performed for that critical region.

2. Phase II - A local finite element elastic-plastic model was analyzed using the DYPLAS computer program which is capable of treating non-linear inelastic materials.

   The boundaries of this model (as shown in Figure 3.8-62 are structurally remote from the jet impinged area, indicated by cross-hatching. The displacements from the general shell analysis therefore were used as displacement boundary conditions. Inelastic deformation, strains and stresses were computed for all finite elements during selected steps of load applications.

3. Stress evaluation was based on the ASME, Boiler and Pressure Vessel Code, Section III, Subsection NE, Class MC Components, NE-3131.2 and Appendix F, 1974 edition.

   3.1 Code Requirements

   $P_m < \sigma_m$

   $P_L < 1.5\ \sigma_m$

   Where:

   $\sigma_m = 0.85 \times 0.7 \times S_u = 0.595\ S_u$

   From Table I-1.1, for SA-516, Grade 70 Steel:

   $S_u = 70{,}000$ psi at $345^{\circ}F$

   Therefore:

   $\sigma_m = 41{,}650$ psi

   To meet code requirements:

   $P_m < 41{,}650$ psi

   $P_L < 62{,}475$ psi

   3.2 Stress Evaluation

   From Appendix J of Reference 3.8-21

   $(P_m)_{max} = 29{,}043 \text{ psi} < 41{,}650 \text{ psi}$

which occurs on upper edge of model (Figure 3.8-62)

$(P_L)_{max} = 36{,}305$ psi $< 62{,}475$ psi

which occurs on the lower portion of the impinged region (Figure 3.8-62)

Therefore code requirements are met.

The ductility ratio is defined as ~~The~~ the maximum response of an elasto-plastic structure to a prescribed loading function divided by the response of the same structure to the load at incipient plasticity. The maximum radial displacement at incipient plasticity was computed, and is shown in computer Appendix I to Reference 3.8-21 as .4335 inches. The maximum radial deflection of 2.0 inches was the spacing between shell and shield.

Thus the ductility ratio might be considered to be:

$$2.00/.4335 = 4.6$$

However, since the maximum deflection of the shell is limited by contact with the concrete of the biological shield; the ductility ratio here may not be as meaningful a design parameter as it is in other cases.

A more significant measure of structural integrity against cracking and ultimate leakage may be the most severe local principal cumulative strain.

The most severely strained metal lies on the outside surface of the vessel shell near node 67 as shown in Figure 3.8-63. This strain is the maximum inelastic cumulative strain based on a computed non-linear strain distribution through the vessel wall. Its value is 4%.

Figure 3.8-63 shows the minimum elongation to failure of ASTM A516 Grade 70 steel as a function of temperature. Over the range of 70°F to 350°F this has a least value of: 14.8%.

An estimate of the factor of safety may then be said to be:

$$F.S. = \frac{14.8}{4.0} = 3.7$$

Although failure strains in complex strain fields do not correlate precisely with results of one-dimensional ductility tests; the factor of safety, as computed above, is deemed ample.

The criteria used in the seismic design of the containment system is based on the responses of the containment system to earthquake excitation. The responses are derived from the analysis of a mathematical model developed to represent the containment vessel. Section 3.7 outlines methods of analysis, modeling techniques, the seismic input and the soil-structure interaction effects.

All ferrous materials of plates, forgings, castings and pipes for ASME Code, Class MC with thickness greater than 5/8 inch are Charpy V-notch impact tested at 0°F in accordance with the ASME Code, Section III, Paragraph NE-2300. Materials for ASME Code, Class I components are impact tested in accordance with NB-2300. The drywell will not be pressurized or subjected to substantial stress at temperatures below 30°F.

The principle containment design parameters are listed in Table 3.8-1.

The physical dimensions of the steel primary containment vessel are:

a. The diameter of the cylindrical portion at the base of the cone is approximately 86 ft.

b. The diameter at the top of the cone is approximately 39.5 ft. and then narrows to 32 ft. to carry the removable head.

c. Ellipsoidal bottom head with a ratio of 2:1 has an inside height of approximately 21.5 ft.

d. The removable ellipsoidal top closure head has an inside height of approximately 15.5 ft.

3.8-19 Shannon and Wilson, Inc., Soil Compaction Evaluation of Quality Class I Backfill, Washington Public Power Supply System, WPPSS Nuclear Project No. 2 (WNP-2).

3.8-20 Letter GO2-78-45, D. L. Renberger to S. A. Varga, entitled WPPSS Nuclear Project No. 2, transmitting an update (Revision 1) to the report in reference 3.8-15, dated February 2, 1978, Docket No. 50-397.

3.8-21 "Primary Containment Vessel for Washington Public Power Supply System, Hanford No. 2, Jet Impingement Analysis," FIRL Technical Report F-C4121, May 21, 1975.

3.8-22 "HYBOS," FIRL Users Manual, July 1973.

Q. 130.034
(3.8.2)

In 3.8.2.2 of the FSAR, you state that the structural steel attachments beyond the boundaries established for the primary containment steel vessel are designed and constructed according to AISC specifications for the design of structural steel buildings where applicable. You further state that you use allowable stress limits in accordance with Sub-Article NE-3131 (3) of Section III of the ASME Code. Identify these structural steel attachments and provide your basis for applying two different structural design codes for the design of the same structural elements.

Response:

Please see revised 3.8.2.2.*

*Draft FSAR revised page changes attached.

3.8.2.2.3.4 Attachments

Structural steel attachments beyond the boundaries established for the steel primary containment vessel are designed and constructed according to the AISC Specification for the Design, Fabrication and Erection of Structural Steel for Building, February 12, 1969, where applicable. ~~The allowable stress limits are in accordance with Sub-Article NE-3131 (3) of Section III of the ASME Code.~~ Non-pressure vessel elements such as catwalks and interior beam connections with hatch floors are designed in accordance with the AISC Code. Stress intensity limits are in accordance with the allowables permitted by the AISC Code with the exception that, for loading combinations including ~~1/2 SSE~~ OBE, no increase in allowable stress will be permitted.

Supports for pipe whip guide rings are designed as described in 3.6.

3.8.2.2.4 Conformance with Regulatory Guides

The following regulatory guides related to the primary containment vessel are applicable to WNP-2 and their implementation is herein discussed.

3.8.2.2.4.1 Regulatory Guide 1.7, Rev. 1 - Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident

Two hydrogen recombiners are provided outside the primary containment vessel, to control the hydrogen buildup following a loss-of-coolant accident as specified by Regulatory Guide 1.7, Rev. 1.

3.8.2.2.4.2 Regulatory Guide 1.11, Rev. 0 - Instrument Lines Penetrating Primary Reactor Containment

For instrument sensing lines penetrating primary containment the following is provided and, accordingly, implements Regulatory Guide 1.11, Rev. 0:

The inner and outer skirts for the support of the primary containment vessel are designed in accordance with Article NE-3100 of Section III of the ASME Code.

Q. 130.035
(3.8.2)

The load combinations considered in your analysis appear to be different than those delineated in Section 3.8.2 of the SRP. Provide justification for these apparent deviations and/or provide an assessment of the significance of the deviations with respect to the adequacy of the structural design. Indicate how a particular type of structural load (e.g., moments, shear, or stresses) resulting from a number of different loads, were combined (i.e., either by using the absolute sum or by the SRSS methodology).

Response:

The loading combinations given in FSAR 3.8.2.3.12 cover all the loading combinations given in Section 3.8.2 of the SRP. Table I, enclosed, provides the loading combinations given in 3.8.2.3.12. Table II, enclosed, presents the correlations between the loading combinations given in the SRP and in 3.8.2.3.12 of the FSAR. The definitions for the loads, appearing in Table I, are the same as those in the SRP Section 3.8.2.II.3. All the loading combinations are performed by using the absolute sum methodology.

TABLE I

FSAR LOADING COMBINATIONS (Notations are defined in SRP 3.8.2)

FSAR LOADING CONDITIONS	D	L	F_L	T	R See Note c	P		Y		E
a) Initial Proof Load Test Condition	D	L		T_t See Note a	See Note f	P_t				~~E~~
b) Final Proof Load Test Condition	D	L See Note h		T_t See Note a	See Note f	P_t	P_e			~~E~~
c) Normal Operating Condition (+OBE/SSE)	D	L See Notes h, i		T_o See Note b	See Note f		P_e			E/E^1
d) Refueling Condition (+OBE)	D	L See Notes h, i		T_o See Note b			P_e			E
e) Accident Condition (+OBE/SSE)	D	L See Notes h, i		T_a ~~T_a~~	See Notes f, j	P_a ~~P_a~~ See Note k	P_e		Y_j	E/E^1
f) Accident Condition (+Pipewhip +OBE/SSE)	D	L See Notes h, i		T_a ~~T_a~~	See Note f, j	P_a ~~P_a~~ See Note k	P_e	Y_m See Note e	Y_j	E/E^1
g) Normal Condition (+Pipewhip +OBE/SSE)	D	L See Notes h, i		T_o See Note b	See Note f		P_e		Y_r	E/E^1
h) Flooded Condition (+OBE)	D	L See Note i	F_L See Note g				P_e			E

NOTES:

a. T_t (the test temperature) is ambient and gives negligible test temperature loads.

b. The normal condition for the thermal loads discussed in 3.8.2.3.12 ~~in the FSAR~~ includes the startup and shutdown transients.

c. All appropriate pipe reactions due to the various loading conditions are included in the stress analysis of the primary containment vessel, although they are not specifically listed in ~~the FSAR.~~ 3.8.2.3.12.

d. T_e, the thermal loads during a thermal event causing external pressure is never greater than T_o for WNP-2.

e. Y_m ~~(to be provided at a later date.)~~ The missile impact equivalent static loads, are not applicable to WNP-2 as no credible missiles impacting the containment have been identified. If the current pipe break and missile study inside containment results in postulated missiles striking the containment, a structural analysis of containment will be performed and the results reported in a future Amendment to the FSAR. (Refer also to Section 3.5).

f. Jet blowdown reactions are ~~included in the FSAR.~~ described in 3.8.2.3.12.

g. F_L, the loads due to post LOCA flooding of containment includes all other applicable reactions, although they are not specifically listed as F_L in the FSAR.

h. The dry well refueling bellows loads ~~will be~~ are considered as live loads.

i. The floor seal loads are considered as live loads.

j. The containment spray header jet nozzle loads are considered as the reaction loads, R.

k. ~~The post accident internal design pressure~~ Pe, is ~~for the containment vessel is negative 2 psig. There is also an~~ External pressure, ~~of an additional~~ 2 psig due to the filler material. ~~FSAR 3.8.2.3.12.b(4). The total post accident external design pressure is the sum of the above two loads and is equal to 4 psig.~~ Internal pressure, Pa, is 45 psig decaying to −2 psig (Refer to 3.8.2.3.12 e and f.)

TABLE II

SRP & FSAR LOADING COMBINATIONS

SRP LOADING CONDITION	CORRESPONDING LOADING	FSAR CONDITION
1) Test	(a)	Initial Proof Load Test Condition
1) Test	(b)	Final Proof Load Test Condition
(2) Normal Operating	(c)	Normal Operating Condition (+OBE/SSE)
(3) Normal Operating (+OBE)	(c)	Normal Operating Condition (+OBE/SSE)
-	(d)	Refueling Condition (+OBE)
(4) Accident Condition (+OBE)	(e)	Accident Condition (+OBE/SSE)
(5) External Loading Condition (+OBE)	(e)	Accident Condition (+OBE/SSE)
(6) Accident Condition (+SSE)	(e)	Accident Condition(+OBE/SSE)
(7) External Loading Condition (+SSE)	(e)	Accident Condition (+OBE/SSE)
(8) Accident Condition (+SSE+Pipe-whip)	(f)	Accident Condition (+Pipewhip + OBE/SSE)
-	(g)	Normal Condition (+Pipewhip + OBE/SSE)
(9) Flooded Condition (+OBE)	(h)	Flooded Condition (+OBE)

Q. 130.036
(3.8.2)

The computer programs AX1, AX2, and AX3 which you cite in 3.8.2.4.2 of the FSAR, are for analysis of axisymmetric structures. Explain how these programs incorporate the effect of the vertical and horizontal stiffeners which reinforce the primary containment vessel.

Response:

Please refer to revised 3.8.2.4.2.*

*Draft FSAR revised page changes attached.

a. AX1, Analysis of Axisymmetric Solids

b. AX2, Axisymmetric Shell Program

c. AX3, Analysis of Thin-Shell Solids of Revolution

The AX1 computer program is a special purpose finite element program for the analysis of axisymmetric solids. Meridional-stiffening cannot be modeled in this program. Circumferential stiffening can be modeled with the axisymmetric finite elements. This program was used exclusively to evaluate jet load effects on the CRD Removal Hatch. All the members analyzed are unstiffened axisymmetric solids. It is capable of determining deformations and stresses within axisymmetric structures of arbitrary shape.

The AX2 computer program is a general purpose program for the analysis of composite shells of revolution loaded axisymmetrically. The shell theory used is for thin, isotropic, elastic shells. Meridional stiffening cannot be modeled. Circumferential stiffening members can be modeled as shell elements or as concentrated stiffnesses at node points. This program was used extensively in the analysis of the primary containment vessel. Meridional stiffening members were neglected in these AX2 analyses. Circumferential stiffening members were modeled as shell elements. It is capable of determining stresses and displacements in shells of revolution that are loaded axisymmetrically.

The AX3 computer program is a general purpose program for the analysis of composite axisymmetric shells. Loads can be applied axisymmetrically or non-axisymmetrically using fourier series representations. Meridional stiffening can be modeled by adding shell layers with zero modulus of elasticity in the circumferential direction. By adding a layer or layers of appropriate thickness(es) the axial and bending stiffness of the stiffening can be represented. Circumferential stiffening members can be modeled using shell elements or by using the layer method described above. It is capable of determining reactions and deformations for thin shell solids of revolution, such as discs, for axisymmetric loads.

The AX3 program was used to analyze two areas:

1. The containment shell in the vicinity of the seismic stabilizers (El. 567'-5-½") under a non-axisymmetric displacement loading. No meridional or circumferential stiffening exists in the region modeled.

2. The Equipment Hatch head/flange intersection under Jet loading. Meridional stiffening on the Equipment Hatch head and barrel was modeled using additional shell layers with zero modulus in the circumferential direction. No circumferential stiffening members exist in the area modeled.

Computer programs AX1, AX2 and AX3 are further discussed in 3.12.

3.8.2.4.3 Seismic Analysis

The evaluation of the structural integrity of the steel primary containment vessel, when excited by seismic motion, is based on a dynamic analysis.

The steel primary containment vessel is designed to interact as a structural component with the reactor building (secondary containment structure) to which it is attached. The primary containment vessel is attached to the reactor building at the stabilizer truss level through the primary containment vessel/ biological shield wall shear lug interface, and at the reactor building foundation mat level.

The structural components within the steel primary containment vessel, such as the reactor pedestal, reactor vessel and sacrificial shield wall are designed to interact with the reactor building and the primary containment vessel because of their connections at the reactor building foundation mat level, drywell floor level, and the top of the sacrificial shield wall level, through the reactor vessel shear lug/ shear lug stabilizer/ sacrificial shield wall stabilizer truss interfaces.

Q. 130.037
(3.8.2)

Provide in 3.8.2.5 of the FSAR, a detailed presentation of your buckling criteria, including your procedures to establish these criteria, for the primary containment vessel.

Response:

Please refer to revised 3.8.2.5.4.*

*Draft FSAR revised page changes attached.

accident condition tests include exposure to steam and containment water spray solutions under temperature-time conditions which are more severe than those that would be encountered in a design basis accident.

All exterior surfaces of the steel primary containment vessel shell are cleaned in accordance with the requirements of Steel Structures Painting Council Surface Preparation Specification No. 6, Commercial Blast Cleaning, latest edition. After cleaning and after having passed inspection, one prime coat of Amercoat Corporation Dimetcote No. 6, minimum 3.0 mils thick, is applied.

All interior surfaces of the primary containment vessel shell and metal surfaces of attachments thereto, except those parts embedded in the base slab, are given the following protective coatings:

a. The drywell: one prime coat of Ameron Corporation Dimetcote No. 6 topcoated with one coat of Amercoat 90 modified phenolic epoxy coating. Surfaces receive an SP-10 surface preparation.

b. The non-immersion surfaces of the suppression chamber: one coat of Dimetcote No. 6. Surfaces receive an SP-10 surface preparation.

c. The immersion surfaces of the suppression chamber: two coats of Amercoat 90, 10 to 14 mil dry film thickness. Immersion surfaces receive an SP-5 surface preparation.

3.8.2.5 Structural Acceptance Criteria

The structural acceptance criteria for the steel primary containment vessel, namely, the basis for establishing allowable stress values, the deformation limits, and the factors of safety, are established by and in accordance with the ASME Code, Section III.

In addition to the structural acceptance criteria, the steel primary containment vessel is designed to meet minimum leakage rate requirements. The leakage rate requirements are discussed in Chapter 6.

The buckling analysis of the containment vessel was performed as follows:

External Pressure - The allowable working pressure, Pa, calculated in NE-3133.3 was compared with the specified maximum

Insert on Page 3.8-46

external pressure, -4 psi. Conical shell elements were analyzed as equivalent cylinders in accordance with NE-3133.7.

Longitudinal Compression on Unstiffened Shell - The maximum allowable compressive stress, B, determined in NE-3133.6 was compared to the maximum longitudinal compressive stress produced under all the loading conditions specified, including the compressive stress due to the SSE overturning moment.

Longitudinal Compression on Meridionally Stiffened Shell - Two independent checks were made on buckling of stiffened shell lengths:

1. NE-3133.6 was applied as above using an equivalent thickness in bending, $t_e = (12 \times I_s/b)^{1/3}$

   b = meridional stiffener spacing

   I_s ~~Is~~ = moment of inertia of the composite section comprised of the stiffener and a width ~~os~~ of shell, b.

2. Additionally, shell lengths were analyzed by treating the composite stiffener - shell described in (1.) above as a column pinned at the shell ring stiffeners. These columns were evaluated for buckling using the AISC criteria.

~~Longitudinal Compression on Conical Shell Lengths - Conical shell lengths were analyzed as described above. To apply NE-3133.6, an equivalent cylindrical radius was computed as:~~ Insert on Page 3.8-46

$$\cancel{R = \frac{\text{Maximum Horizontal Radius}}{\text{Cos (Cone Angle = 16.05}^{\circ})}}$$

3.8.2.5.1 Stress Limits for Design Loading Conditions

The requirements of Section 3000 of the ASME Code, Section III, as modified by Regulatory Guide 1.57, Rev. 0 and discussed in 3.8.2.2.4.6, are met for each of the load

combinations prescribed in 3.8.2.3.12, Loading Combinations. This is discussed in greater detail in 3.8.2.4, Design and Analysis Procedures. Buckling criteria is discussed in 3.8.2.5.4, Buckling Criteria for the Primary Containment Vessel.

3.8.2.5.2 Primary and Secondary Stresses

For loading combinations described in 3.8.2.3.12a through 3.8.2.3.12e, inclusive and 3.8.2.3.12g, the stress limits specified in ASME Code, Section III, NE 3131 (c) are utilized.

3.8.2.5.3 Peak Stresses

For loading combinations described in 3.8.2.3.12f and 3.8.2.3.12h, the stress limits specified in the ASME Code, Section III, NE 3131(c)(1) and NE 3131(c)(2) are utilized.

3.8.2.5.4 Buckling Criteria for the Primary Containment Vessel

To assure safety against buckling, the rules set forth in the ASME Code, Section III, NE 3133 are utilized. ← [INSERT]

3.8.2.6 Materials, Quality Control and Special Construction Techniques

3.8.2.6.1 Materials

The material essential to containment integrity comply with the requirements of Article NE-2000 of NE of the ASME Code, Section III. Material for the containment vessel (3.8.2) and vessel penetrations (3.8.6) conforms to , but is not limited to, the following:

a. Plate:

ASME SA-516, Gr. 70 (Drywell head, flanges, containment vessel and electrical penetration weld ring collars)	Specification for Carbon Steel Plates for Pressure Vessels for Moderate and Lower Temperature Service.

Q. 130.038
(3.8.3)

References 3.8-5, 3.8-6 and 3.8-7 of the FSAR have not been submitted for our review and acceptance. It is our position that the use of reports or references for definitions, criteria, and methods of analysis is unacceptable until the referenced documents are reviewed and accepted by us. Accordingly, submit the cited references for our review. Alternatively, refer to other documents which have been previously accepted by us.

Response:

Reference 5 was submitted to the NRC by WPPSS letter GC2-74-41, dated March 21, 1974, as a response to a post-construction permit item identified in a meeting at the NRC.* Reference 6 was submitted to the NRC by WPPSS letter G02-75-181, dated June 26, 1975, in response to NRC further requests for information. In the same light, Reference 7 was submitted to the NRC on WPPSS letter G02-75-240, dated August 19, 1975. NRC to WPPSS letters (De Young to Stein), dated August 13, 1975 and October 15, 1975, documented that the submitted information was acceptable.

*Letter, W. R. Butler (NRC) to J. J. Stein (WPPSS), transmitting minutes of the October 17-18, 1973 meeting on post-construction permit items, meeting agenda item 10, dated November 20, 1973.

Q. 130.039
(3.8.3)

Table 3.8-10 of the FSAR presents the load combinations and load factors for internal structures made of reinforced concrete. However, it appears from this table that the abnormal/severe environmental loads were not considered. Similarly, it appears from our review of Table 3.8-11 that the extreme environmental, abnormal, and abnormal/severe environmental loads were not considered for internal steel structures. Additionally, it appears that live loads were not considered in some of the load combinations in this latter table. Accordingly, provide justification for these omissions and/or assess the significance of these ommissions with respect to the adequacy of the WNP-2 structural design.

Response:

The question indicates that certain loadings and load combinations in current NRC criteria for concrete and steel structures internal to the steel containment have not been considered in the design; justification is requested. In the response, comparison is made between the actual criteria established for design and current NRC criteria listed in the Standard Review Plan. Justification for the actual design and criteria is also presented in the response. Design bases for these structures were previously submitted to and approved by the NRC in 1975 docket letters referenced.

Concrete Structures

The pertinent concrete structures internal to the steel containment are the drywell floor slab, the wetwell columns and the reactor pedestal. FSAR Table 3.8-10 lists the load combinations which were used for all internal concrete structures except the reactor pedestal. The load combinations used for the reactor pedestal are listed in FSAR Reference 3.8-6 which was submitted to NRC by WPPSS letter G02-75-37, dated 2/11/75 and approved by NRC by letter dated 8/13/75, Docket No. 50-397 (with additional information being submitted on WPPSS to NRC letter G02-75-181, dated 6/26/75).

The question states that the combination involving abnormal/severe environmental loads was not considered. A comparison of the pertinent load combinations used in design and those in Standard Review Plan (SRP) 3.8.3 is given in Table Q. 130.39-1. Symbols in the FSAR are used.

TABLE Q. 130-39-1

INTERNAL REINFORCED CONCRETE STRUCTURES - PERTINENT LOAD COMBINATION

COMBINATION 4 - ABNORMAL CONDITIONS

SRP:	$D + L + T_a + 1.5\ P_a + R_a$
TABLE 3.8-10:	- - -
Ref. 3.8-6	$D + L + T_a + 1.5\ P_a + R_a$

COMBINATION 5 - ABNORMAL/SEVERE ENVIRONMENTAL CONDITIONS

SRP:	$D + L + T_a + 1.25\ P_a + 1.0\ R_r + 1.25\ E$
TABLE 3.8-10:	- - -
Ref. 3.8-6	$D + L + T_a + 1.25\ P_a + 1.0\ R_r + 1.25\ E + R_a$

COMBINATION 6 - ABNORMAL/EXTREME ENVIRONMENTAL CONDITIONS

SRP:	$D + L + T_a + 1.0\ P_a + 1.0\ R_r + 1.0\ E' + R_a$
TABLE 3.8-10:	$D + L + T_a + 1.0\ P_a + 1.0\ R_r + 1.0\ E' + R_a$
Ref. 3.9-6:	$D + L + T_a + 1.0\ P_a + 1.0\ R_r + 1.0\ E' + R_a$

Review of the table shows that, for the reactor pedestal (Reference 3.8-6), the SRP combinations have been satisfied.

In regard to the other concrete structures internal to containment, explicit consideration of the abnormal/severe environmental condition (Combination 5) is not included in Table 3.8-10. However, the abnormal/extreme environmental condition (Combination 6) of SRP has been complied with identically. By utilizing a conservative relation between E' and E characteristic of WNP-2, namely $E' \geq 1.6\ E$, in Combination 6, the equivalent Combination 5 under abnormal/severe environmental conditions is obtained. This is listed below together with the SRP combination.

SRP Combination 5: $D + L + T_a + 1.25\ P_a + 1.0\ R_r + 1.25\ E$

Table 3.8-10 Equiv.
Combination 5: $D + L + T_a + 1.0\ P_a + 1.0\ R_r + 1.6\ E + R_a$

It is seen from the above that a load combination involving abnormal/severe environmental loads was considered implicitly. In comparison with SRP Combination 5 the equivalent Combination 5 has a larger load factor for E but a smaller load factor for P_a.

It is further noted with regard to these concrete structures that the combination involving abnormal conditions (Combination 4) is also not included in Table 3.8-10. However, comparison with the included Combination 6 shows that the generic load terms of Combination 4 have been considered, but with different load factors. Thus, Combination 4 includes the term 1.5 P_a as aginst 1.0 P_a in Combination 6, but Combination 4 omits the term 1.0 R_r and 1.0 E' which are present in Combination 6.

As previously noted, the internal concrete structures whose decigns were based on the load combinations of FSAR Table 3.8-10 are the drywell floor slab and the wetwell columns. The designs of these structures have been checked for compliance with SRP Combinations 4 and 5 and found to satisfy these combinations. Pertinent information is noted below.

a. Drywell Floor Slab - The principal load controlling slab design is the postulated pipe break jet impingement force (R_r). Other loads such as live load (L) and a 25 and 50 percent increases in Combinations 5 and 4 in the differential pressure on the slab (P_a) each cause less than 2 percent of the design bending moment; the effect of the seismic load (E') is still smaller. Analysis shows that the drywell floor slab has adequate capacity to sustain SRP Combinations 4, 5 and 6.

b. Wetwell Column - For: Combination 5, the increase in axial load due to the 25 percent increase in differential pressure on the drywell floor is less than 10 percent of the design axial load. The design bending moment is reduced by 50 percent as OBE is used in the load combination instead of SSE. The resultant combination of loads is within the capacity of the column.

Comparing Combination 4 with Combination 6, the axial load due to differential pressure is increased by 50 percent but it decreases due to the omission of loads caused by the safe shutdown earthquake (E') and pipe break jet force (R_r). The overall effect is a decrease in design axial load. The concurrent bending moment in Combination 4 vanishes as no seismic

load is included. The resultant combination of loads is within the capacity of the column. Thus, Combinations 4 and 5 as well as Combination 6 are satisfied.

Steel Structures

The pertinent steel structures internal to the steel containment and the sacrificial shield wall, the stabilizer truss, the drywell floor beams, and the radial beam framing systems. FSAR Table 3.8-11 lists the load combinations which were used for all internal steel structures except the sacrificial shield wall. The load combinations used for the sacrificial shield wall are listed in FSAR Reference 3.8-7 which was submitted to NRC by WPPSS letter GO-75-240 dated 8/19/~~79~~ and 75 approved by NRC by letter dated 10/15/75, Docket No. 50-397. In design, the elastic working stress design method of Part 1 AISC, 1969 was used with the associated load combinations. FSAR Table 3.8-11 also lists the load combinations applicable to the plastic design method, but these were not used and were included only for the purpose of having an overall complete table of load combinations.

The question states that three load combinations were not considered, namely, those for extreme environmental conditions, abnormal conditions, and abnormal/severe environmental conditions. It further notes that live loads were not considered in some of the load combinations. Justification and/or assessment of the omissions is requested.

A comparison of the combinations for factored load conditions as used in design and as in Standard Review Plan 3.8.3 is given in Table Q. 130.39-2. Symbols in the FSAR are used. As stated above, the elastic stress design method of Part 1, AISC, 1969 was used in design.

TABLE Q. 130-39-2

INTERNAL STEEL STRUCTURES - PERTINENT LOAD COMBINATIONS

COMBINATION 3 - EXTREME ENVIRONMENTAL CONDITIONS

SRP:	$D + L + T_o + R_o + E'$
TABLE 3.8-11:	- - -
Ref. 3.8-7	$D + L + T_o + R_o + E'$

COMBINATION 4 - ABNORMAL CONDITIONS

SRP:	$D + L + T_a + R_a + P_a$
TABLE 3.8-11:	- - -
Ref. 3.8-7	$D + L + T_a + R_a + P_a$

COMBINATION 5 - ABNORMAL/SEVERE ENVIRONMENTAL CONDITIONS

SRP:	$D + L + T_a + R_a + P_a + R_r + E$
TABLE 3.8-11:	- - -
Ref. 3.8-7:	$D + L + T_a + R_a + P_a + R_r + E$

COMBINATION 6 - ABNORMAL/EXTREME ENVIRONMENTAL CONDITIONS

SRP:	$D + L + T_a + R_a + P_a + R_r + E'$
TABLE 3.8-11:	$D + T_a + R_a + P_a + R_r + E'$
Ref. 3.8-7:	$D + L + T_a + R_a + P_a + R_r + E'$

Review of the table shows that for the sacrificial shield wall the SRP combinations have been satisfied.

The load combinations in FSAR Table 3.8-11 are in accord with criteria current at the time of design of the applicable internal structures. It is noted that SRP Combinations 3, 4 and 5 reflecting extreme environmental, abnormal, and abnormal/severe environmental loads respectively were not included in the table. This is due to the fact that NRC documents listing these combinations were issued subsequent to completion of design. However, the table does provide for the

aforementioned loads but not in the same combinations as the SRP.

In regard to the stabilizer truss and the drywell floor beams the following is pertinent:

a. The major loads controlling the design of the stabilizer truss are the safe shutdown earthquake (E'), and the results of pipe break including pressurization (P_a) and local pipe break forces (R_r).

b. The major loads controlling the design of the drywell floor beams result from pipe break and include pressurization (P_a) and jet force (R_r).

c. FSAR load Combination 6 including the above major loads was used in the design of the structures. The effect on the design of the live load (L) associated with Combination 6 is minor. The current designs have sufficient margin to sustain the required loads including the additional live load as in SRP load Combination 6.

d. The allowable stresses actually used in design with load Combination 6 are less than ~~or equal to~~ 1.6 S. The loading of Combination 6 exceeds the loadings of Combinations 3, 4 and 5. The design stress level allowed in the SRP for Combinations 3, 4 and 5 is 1.6 S. Hence, Combination 6, as actually used, controls with respect to Combinations 3, 4 and 5.

In view of the preceding, the stabilizer truss and the drywell floor beams have sufficient capacity to sustain the SRP Load Combinations 3, 4, 5 and 6.

For the radial beam framing systems considered in FSAR 3.8.3, the controlling loads used in design involve dead and live loads, reactions under operating conditions, and seismic loads; pipe whip loads do not act on the considered framing systems. The pertinent design load combinations are Combinations 1, 1a, 2, 2a and 6 as listed in FSAR Table 3.8-11.

Analysis and design of radial beam framing systems subjected to pipe whip loads are covered in FSAR 3.6. The structural members which support pipe whip restraints are designed for all applicable load combinations. For normal and occasional loading conditions, the structural members are designed for

load combinations given in FSAR Table 3.8-11. For loading conditions which include the effects of postulated pipe rupture, the structrual members are designed as described in FSAR 3.6.2.2.2 and 3.6.2.3.3.2. In determining the required strength of structural members the normal loads, such as dead load and pipe reaction loads, are superimposed on the dynamically arrived at pipe rupture loads.

As noted in FSAR Table 3.8-11, the design of the radial beam systems did not provide for simultaneous application of live load and seismic forces. This is in accordance with PSAR Table 12.2-2. It is justified on the basis that the live load magnitudes used in Design Load Combinations 1 and 1a represent short-time equipment lay-down loads only possible while the plant is not operating. Such short-time loads are not normally included with seismic loads in current seismic design practice as evidenced in "Recommended Lateral Zone Requirements and Commentary" by the Seismology Committee of the Structural Engineers Association of California. A similar rationale was considered for load combinations involving live load in conjunction with seismic and pipe rupture loads.

Q. 130.040
(3.8.3)

Though Section 3.8.3.4 of the FSAR indicates that it contains a description of the methods of analysis for internal structures, we cannot find any such description in this section. Accordingly, provide a discussion of your method of analysis for computing forces and displacements in all internal structures. If computer codes were used for this analysis, identify these codes. Provide or describe, whenever applicable, the models used in performing such analyses. Provide comparisons between the design values of forces and displacements and their allowable values based on: (1) the allowable stress limits; or (2) the capacity determined by the ultimate strength of these structures.

Response:

The question apparently indicates that the description of the methods of analysis in 3.8.3.4 covering internal structures is inadequate. In the response, the design and analysis procedures are rewritten and clarified as appropriate. See revised 3.8.3.4.*

*Draft FSAR revised page changes are attached.

e. Temperature and pressure effects during normal operating and accident conditions.

Internal structures are designed for the reactions of all other structures or equipment that they may support including the steam supply system hangers and supports.

The reactor vessel stabilizer truss is designed primarily for lateral seismic loads. However, all the loads associated with a support at the top of the sacrificial shield wall such as pressure and pipe whip loads on the sacrificial shield wall are included in the design of the stabilizer truss. The applicable load combinations are found in Table 3.8-11.

3.8.3.4 Design and Analysis Procedures

3.8.3.4.1 Reactor Pedestal

The general approach in the analysis and design of the reactor pedestal is to determine the values of the controlling stress resultants on the basis of elastic analysis under design loadings and to provide the required capacity of the pedestal in accordance with the strength method of the ACI 318-71 Code (Reference 3.8-10). Design loadings are in conformance with the loads and load combinations of 3.8.3.3. The report on design and analysis procedures for the upper portion of the pedestal, including transmission into the pedestal of reactions from the sacrificial shield wall and the reactor pressure vessel, is contained in Reference 3.8-6 which was submitted to NRC by WPPSS letter G02-75-36 dated 2/11/75 and approved by NRC by letter dated 8/13/75, Docket No. 50-397.

The principal loadings controlling the design of the pedestal are due to seismic action and pipe break effects which include annulus pressurization, pipe reactions and pipe whip forces. Controlling overall stress resultants are pedestal bending moment, shear and axial force. The values of these stress resultants due to seismic action are obtained from a dynamic analysis of a discrete mathematical idealization of the entire reactor building structure as described in 3.7. For other loadings, the pedestal is analyzed as a cylindical beam fixed at its base and simply supported at the level of the drywell floor.

The distribution over the cross section of the axial stresses and shearing stresses due to the aforementioned stress resultants is that associated with single flexural theory. The axial force per unit length of arc due to the overall bending moment varies lineary with the distance from the neutral axis. The shearing force per unit length of arc is

circumferential (tangential) in direction and varies sinusoidally in magnitude with maximum at the neutral axis. Meridional and hoop steel requirements are determined at the locations of maximum stress; this reinforcement is then provided uniformly around the pedestal.

Specific analyses are made at pedestal discontinuities such as openings and boundaries to determine the radial shears and radial moments. Additional reinforcement consisting of radial ties and meridional steel is provided as required.

At the base of the pedestal, provision is made for transmission of the pedestal reactions. Capacity for the transmission of shear is available due to axial compression, shear friction and the continuous key of the pedestal into its base. The axial tensite forces are transmitted by continuing the meridional reinforcement into the base where the connection to the cadweld ring assembly is made. This assembly, in turn, connects through weldments to the inner steel skirt which is anchored to the basemats.

3.8.3.4.2 Sacrificial Shield Wall

The design and analysis procedures for the sacrificial shield wall are described in Reference 3.8-7 which was submitted to NRC by WPPSS letter G02-75-240 dated 8/19/75 and approved by NRC by letter dated 10/15/75, Docket No. 50-397.

3.8.3.4.3 Drywell Floor Structural System and Support Elements, Including the Peripheral Seal Assembly, Peripheral Seal Jet Deflectors, and Peripheral Shear Lugs

3.8.3.4.3.1 Drywell Floor Slab and Columns

The drywell floor slab and columns are each analyzed elastically to determine the values of controlling stress resultants under the design load combinations for concrete structures as in 3.8.3.3. The required capacity for each of these structures is then provided in accordance with the strength method of the ACI 318-71 Code (Reference 3.8-10).

Under vertical loading, the floor slab is considered to act as a one-way slab in the radial direction, supported by tangential beams below, and extending from the support at the pedestal to the face end at the primary containment vessel. The slab is analyzed as continuous over the supporting beams except for the spans between downcomers which are taken to be simple spans. Significant loads include dead load and differential pressure on the slab (P_a) but the principal load controlling slab design is the pipe break jet impingement force (R_r). Slab capacity is provided to resist the calculated design shears and moments.

The floor slab is also analyzed for the effect of other significant loads besides vertical loads. The effect of differential temperature between drywell and wetwell including slab bending is checked. Also, the connection between the pedestal and the floor slab is checked for capacity to transmit the pedestal horizontal seismic reaction.

The wetwell columns are subjected to a combination of axial (vertical) and lateral (horizontal) loading resulting from the superimposed loads from the drywell floor and seismic action respectively. The significant loads from the drywell floor, which contribute to the design axial load, are the floor dead load, live load, vertical seismic load (E^z), differential pressure, and jet impingement. Column flexure due to horizontal seismic action is determined from analysis of the column as an elastic member subjected to lateral inertial forces

corresponding to the column seismic acceleration. The capacity of the column to sustain the design stress resultants is determined by the strength method of the ACI 318-71 Code. With concurrent axial load and bending moment, the magnified moment due to axial load is utilized in conformance with the Code.

3.8.3.4.3.2 Structural Steel Members

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The steel beam structural system of the drywell floor consists of a grid of radial and tangential beams which support the drywell floor slab. The ~~24~~ radial beams, which divide the floor area into ~~24~~ similar sectors, are supported by the pedestal and the wetwell columns, and extend as cantileaers beyond the columns to the vicinity of the primary containment vessel. In each sector, the tangential beams carry the drywell floor slab and span between the radial beams.

Each of the radial and tangential beams is analyzed by conventional elastic methods as an overhanging or simple span beam, as appropriate, to determine the design moments and shears. Loadings for the radial beams and tangential beams are as discussed above for the wetwell columns and the drywell floor slab. The beams are designed as composite beams in conjunction with the slab above. The elastic working stress design method of the 1969 AISC Design Specification (Ref. 3.8-11) for composite construction is followed.

3.8.3.4.3.3 Drywell Floor Peripheral Seal Assembly

The drywell floor peripheral seal assembly is shown in Figure 3.8-20. The drywell floor peripheral seal is made of steel and is welded to the primary containment vessel and to the underside of the circular closure girder embedded in the drywell floor. It is a 270° segment of a stainless steel pipe in cross-section, circular in plan, and is drained to the floor drain system which is routed to a point outside of primary containment. Design and construction are compatible with primary containment requirements of Class MC components. Assembly of the seal and attachment thereof to both the floor and the primary containment vessel is by means of welding in accordance with the ASME BPV Code, Section III, Class MC. The floor seal is designed to accommodate the maximum vertical and radial differential thermal movements which may occur during plant startup, normal operation and shutdown. It is also designed to withstand, in an elastic manner, the effects associated with a loss-of-coolant accident, including temperature changes and pressure differentials ranging from (+) 25 psig to (-) 6.4 psig, and seismic loads. No other loads are applied to this seal. Jet deflectors are provided at the seal to prevent the direct impingement of a fluid jet force on the seal due to any pipe break. To prevent differential lateral and torsional movements, shear lugs are furnished along the outer periphery of the drywell floor to ensure that movements of the interfacing drywell floor, floor seal and primary containment vessel are in unision during seismic events.

A continuous circular closure girder, which is of structural steel and embedded in the reinforced concrete drywell (or diaphragm) floor along its periphery, is provided. Its basic function is to complete the drywell floor closure. It consists of a cylindrical vertical web plate extending from the bottom of the radial steel beams supporting the drywell floor

slab to the top of the drywell floor slab at elevation 499'-6" and with annular flanges as illustrated in Figure 3.8-20. In addition to its sealing function, the closure girder also provides the means for connecting the drywell floor peripheral seal to the drywell floor, for attaching the male components of the shear lug assembly, and for supporting the concrete floor. The closure girder withstands the design basis accident loads, drywell floor and slab loads, tangential seismic shear loads and loads from the drywell peripheral steal.

The closure girder is designed according to the 1969 AISC Specification and for normal operating load combinations, extreme environmental and the abnormal/extreme environmental loading combinations. Among the loads included in the combinations are design basis accident loads, drywell floor and slab loads, tangential seismic shear loads and loads from the drywell floor seal. The loads are effectively resisted by the girder in flexure, shear, bearing on the concrete slab and in tension by the way of shear stud connectors and embedded structural steel.

3.8.3.4.3.4 Drywell Floor Peripheral Shear Lugs

Thirty six male shear lugs, equally spaced around the drywell floor periphery, transmit horizontal load between the drywell floor and the primary containment vessel. Each of these lugs consist of an assembly of steel plates joined by welding and anchored to the concrete floor slab by stud shear connectors. Transmission of load to the primary containment vessel is via female shear lugs welded to the vessel. The joint beteen male and female lugs affords restraint only in the circumferential direction as relative motion in the vertical and radial directions is permitted.

Analysis and design of the shear lug assembly is in accordance with the elastic working stress design method, Part 1 of the 1969 AISC Design Specification (Ref. 3.8-11). The principal load controlling the design of the lug assembly is the horizontal seismic force transmitted by the drywell floor as determined by the method in 3.7. In line with elastic theory, the distribution of the shear force per unit length of periphery is taken to vary sinusoidally with maxima along the diameter perpendicular to the direction of the overall shear force. The maximum value of the distributed shear force per unit length is used to design the shear lugs.

3.8.3.4.4 Radial Beam Framing Systems

The radial beam framing systems considered herein are those which do not support pipe whip restraints. Analysis and design of those beam systems which do support pipe whip restraints is

covered in 3.6.2.3.3.2 c, d, and e. The analysis and design of the former radial beam systems is in accordance with the elastic working stress design method, Part 1 of the 1969 AISC Design Specification (Ref. 3.8-11). Conventional elastic beam analysis is used. The significant loads in the load combinations are dead, live, reactions under operating conditions, and seismic loads.

3.8.3.4.5 Stabilizer Truss

The stabilizer truss is a pin connected plane truss which transmits horizontal force between the top of the sacrificial shield wall and the primary containment vessel/biological shield wall, as described in 3.8.3.1.5. This transmitted force represents reactions from the sacrificial shield wall and the reactor pressure vessel. The supports for the truss joints at the sacrificial shield wall are fixed at the wall so that two components of reaction (radial and tangential) may occur. At the primary containment vessel the truss joint support is constrained only in the circumferential direction so that the only reaction is tangential force.

Analysis and design of the stabilizer truss is in accordance with the elastic working stress design method, Part 1 of the AISC Design Specification (Ref. 3.8-11). The principal loads controlling the design of the truss result from seismic action and pipe break effects including pipe whip, pipe jet, and annulus pressurization. The forces transmitted by the stabilizer truss under these loadings are determined by analysis ~~by the stabilizer truss under these loadings are determined by analysis~~ of the overall structural system from the pedestal to the primary containment vessel including the sacrificial shield wall and the reactor pressure vessel. In this regard, the sacrificial shield wall is modeled as a space frame as described in the report referred to in 3.8.3.4.2 and the reactor pressure vessel as a beam to give the loads transmitted by the stabilizer truss. Analysis of the stabilizer truss as a pin connected plane truss with supports as described above is accomplished using the proprietory computer program "McDonnell-ECI, ICES, STRUDL" which is based on MIT's STRUDL II, Version 2, Update 2 as augmented by McDonnell Douglas Automation Co., St. Louis, Missouri. The computer analysis is described in the report referred to in 3.8.3.4.2.

3.8.3.4.6 Refueling Bellows Seals

Design and analysis procedures for the inner and outer refueling bellows seals are based upon applicable ASME Pressure Vessel Code, Sections II, VIII and IX; the Standards of the Expansion Joint Manufacturers Association, and Interpretation Case Number 1177-7. (See Figure 3.8-27).

3.8.3.4.7 Reactor Steam Supply System Hangers and Supports

Design and analysis procedures for the steam supply system hangers and supports are found in 5.4. Design and analysis procedures for the General Electric stabilizers are based upon applicable ASME pressure vessel codes.

3.8.3.4.8 Reinforced Concrete Lining Inside Bottom Head of Primary Containment Vessel

The design accounts for strains caused by creep, shrinkage, and elastic shortening. The methods and data used for the analysis are based on the applicable codes, standards and specifications in Table 3.8-9 and the results of past experience. (See Figures 3.8-1 and 3.8-18).

The concrete lining is analyzed using elastic methods and designed in accordance with ACI 318-71 by the strength method.

The headed stud shear connectors anchoring the concrete liner to the bottom head of the primary containment vessel are capable of transferring horizontal shear from the concrete internal structures to the bottomm head of the containment vessel, and of resisting relative movements between the

concrete liner and the bottom head of the containment vessel. Refer to the discussion of the reactor pedestal in 3.8.3.1.8 and 3.8.3.4.1.

The analysis and design account for any postulated loading conditions which would cause net uplift at the base of certain reinforced concrete columns.

Uplift on any portion of the pedestal base is transmitted directly into the reactor building foundation mat as discussed in 3.8.3.4.1.

3.8.3.4.9 Downcomer Vent Pipes

The downcomer vent pipes are designed to contain and direct uncondensed drywell steam into the suppression pool, following a pipe break accident. See 3.8.3.1.3 and References 3.8-8, 3.8-15 and 3.8-20 for further description and the design and analysis procedures used.

Stainless steel extension pieces were added to the ends of the downcomers to prevent coating damage from plugs which are installed for the pre-operational 10CFR50 Appendix J containment tests. Downcombers were originally provided with exit flanges for these tests. These flanges were removed because of concern about the applicability of test data taken on prototype downcombers without flanges. The downcomers are designed and constructed in accordance with ASME Section III Class 2 requirements above 1 inch above the circumferential weld joining the stainless steel extension pieces to the bottom of the downcombers. Below this point the downcombers are designed and constructed to ASME Section III Class 3 requirements. The only effect of this code break is to eliminate radiography requirements for the circumferential weld.

3.8.3.5 Structural Acceptance Criteria

3.8.3.5.1 Reinforced Concrete

The maximum permissible stresses and strains used are given in Table 3.8-12. These permissible stresses and strains are used to keep the structures below the range of general yield, both under service load conditions and factored load conditions.

For each of the loading combinations delineated in Table 3.8-10, the required sectional strength of concrete (U) is calculated using the strength design method of ACI 318-71 with the applicable capacity reduction factor.

Q. 130.041
(3.8.3)
(3.8.4)

You indicate in Tables 3.8-13 and 3.8-17 of the FSAR that the plastic section modulus might be used in conjunction with the elastic working stress design method for the factored load combinations. Indicate whether you did this in your structural analysis. If so, provide justification for this approach.

Response:

Table 3.8-13 contains acceptance criteria for steel internal structures of steel containment. In applying the table, the pastic section modulus has been used in conjunction with the elastic working stress design method only in factored load combinations 5 and 6. The use of the plastic section modulus in these two load combinations complies with acceptance criteria in 3.8.3 of the NRC Standard Review Plan (SRP). Table 3.8-13 is modified to reflect this restricted use of the plastic section modulus.*

Table 3.8-17 contains acceptance criteria for applicable steel structures outside primary metal containment. In applying the table, it is intended that the plastic section modulus may be used in conjunction with the elastic working stress design method only in factored load combinations 7 and 8. The use of the plastic section modulus in these two load combinations complies with the acceptance criteria in 3.8.4 of the SRP. Table 3.8-17 is modified accordingly.* In the structural analysis for the applicable steel structures outside primary containment, as listed in 3.8.4.1, the plastic section modulus has not been used in conjunction with the elastic working stress design method.

*Draft FSAR page changes attached.

TABLE 3.8-13

SECTION STRENGTH LIMITS AND SECTION MODULUS FOR STRUCTURAL STEEL INTERNAL STRUCTURES OF STEEL CONTAINMENT

		LOAD CATEGORY	LOAD COMBIN. NO.	STRENGTH LIMIT	SECTION MODULUS OF STEEL SHAPES
ELASTIC WORKING STRESS DESIGN METHOD	SERVICE LOAD CONDITIONS	NORMAL	1	S	ELASTIC SECTION MODULUS IS USED.
		NORMAL	1a	1.5 S	
		SEVERE ENVIRONMENTAL	2	S	
		SEVERE ENVIRONMENTAL	2a	1.5S	
	FACTORED LOAD CONDITIONS	EXTREME ENVIRONMENTAL	3	1.6S	
		ABNORMAL	4	1.6S	
		ABNORMAL/SEVERE ENVIRONMENTAL	5	1.6S	PLASTIC SECTION MODULUS MAY BE USED
		ABNORMAL/EXTREME ENVIRONMENTAL	6	1.7S	
PLASTIC DESIGN METHOD	SERVICE LOAD CONDITIONS	NORMAL	1	Y	PLASTIC SECTION MODULUS IS USED.
		NORMAL	1b	Y	
		SEVERE ENVIRONMENTAL	2	Y	
		SEVERE ENVIRONMENTAL	2b	Y	
	FACTORED LOAD CONDITIONS	EXTREME ENVIRONMENTAL	3	0.9Y	
		ABNORMAL	4	0.9Y	
		ABNORMAL/SEVERE ENVIRONMENTAL	5	0.9Y	
		ABNORMAL/EXTREME ENVIRONMENTAL	6	0.9Y	

NOTES:

1. S IS THE REQUIRED SECTION STRENGTH BASED ON THE ELASTIC DESIGN METHODS AND THE ALLOWABLE STRESSES DEFINED IN PART 1 OF THE AISC "SPECIFICATION FOR THE DESIGN, FABRICATION AND ERECTION OF STRUCTURAL STEEL FOR BUILDINGS", FEB. 12, 1969.

2. Y IS THE SECTION STRENGTH REQUIRED TO RESIST DESIGN LOADS BASED ON PLASTIC DESIGN METHODS IN PART 2 OF THE AISC "SPECIFICATION FOR THE DESIGN, FABRICATION AND ERECTION OF STRUCTURAL STEEL FOR BUILDINGS", FEB. 12, 1969.

3. THIS TABLE APPLIES TO 3.8.3

SECTION STRENGTH LIMITS AND SECTION MODULUS FOR SEISMIC CATEGORY I AND NON-SEISMIC CATEGORY I SAFETY RELATED STEEL STRUCTURES OUTSIDE PRIMARY METAL CONTAINMENT

		LOAD CATEGORY	LOAD COMBINATION No.	STRENGTH LIMIT	SECTION MODULUS OF STEEL SHAPES
ELASTIC WORKING STRESS DESIGN METHOD	SERVICE LOAD CONDITIONS	NORMAL	1	S	ELASTIC SECTION MODULUS IS USED
		NORMAL	1a	1.5 S	
		SEVERE ENVIRONMENTAL	2	S	
		SEVERE ENVIRONMENTAL	2a	1.5 S	
		SEVERE ENVIRONMENTAL	3	S	
		SEVERE ENVIRONMENTAL	3a	1.5 S	
	FACTORED LOAD CONDITIONS	EXTREME ENVIRONMENTAL	4	1.6 S	
		EXTREME ENVIRONMENTAL	5	1.6 S	
		ABNORMAL	6	1.6 S	
		ABNORMAL	U12	1.6 S	
		ABNORMAL/SEVERE ENVIRONMENTAL	7	1.6 S	PLASTIC SECTION MODULUS MAY BE USED
		ABNORMAL/EXTREME ENVIRONMENTAL	8	1.7 S	
		ABNORMAL/EXTREME ENVIRONMENTAL	U13	1.7 S	ELASTIC SECTION MODULUS IS USED
		ABNORMAL/EXTREME ENVIRONMENTAL	U14	1.7 S	
PLASTIC DESIGN METHOD	SERVICE LOAD CONDITIONS	NORMAL	1	Y	PLASTIC SECTION MODULUS IS USED
		NORMAL	1b	Y	
		SEVERE ENVIRONMENTAL	2	Y	
		SEVERE ENVIRONMENTAL	2b	Y	
		SEVERE ENVIRONMENTAL	3	Y	
		SEVERE ENVIRONMENTAL	3b	Y	
	FACTORED LOAD CONDITIONS	EXTREME ENVIRONMENTAL	4	0.9Y	
		EXTREME ENVIRONMENTAL	5	0.9Y	
		ABNORMAL	6	0.9Y	
		ABNORMAL	U12	0.9Y	
		ABNORMAL/SEVERE ENVIRONMENTAL	7	0.9Y	
		ABNORMAL/EXTREME ENVIRONMENTAL	8	0.9Y	
		ABNORMAL/EXTREME ENVIRONMENTAL	U13	0.9Y	
		ABNORMAL/EXTREME ENVIRONMENTAL	U14	0.9Y	

SEE NOTES APPLICABLE TO TABLE ON FOLLOWING PAGE.

Q. 130.042
(3.8.4)

For both steel and reinforced concrete structures, the load combinations which you used are different from those presented in Section 3.8.4 of the SRP. Provide your bases for these deviations and/or provide an assessment of the significance of these deviations with respect to the adequacy of the WNP-2 structural design.

Response:

The load combinations and load factors used in design of reinforced concrete and steel structures outside containment are consistent with acceptable engineering standards that have precedent in design of critical structures in nuclear facilities. Justification for deviations between the criteria used in design and the current NRC criteria provided in the Standard Review Plan is given below:

Concrete Structures

The Seismic Category I concrete structures outside containment are listed in 3.8.4.1 of the FSAR. The concrete components of these structures are considered herein. The load combinations applicable to these concrete structures and components are listed in Table 3.8-15 of the FSAR. The table, as submitted previously was used in the design when the applicable loadings did not involve pipe rupture loads. When pipe rupture loads are applicable, revised Table 3.8-15 of the FSAR lists the load combinations which were used in design.*

A comparison of the load combinations in SRP 3.8.4 and those used in design is given in Tables Q130.42-1a, b and c. Service load conditions are covered in Table Q130.42-1a. Factored load conditions without pipe break and with pipe break are taken up in Tables Q130.42-1b and 1c respectively. The strength design method was used with all load combinations. Symbols in the FSAR are used in the tables.

Review of the aforementioned tables shows that for Factored Load Conditions, both without pipe break and with pipe break (Tables Q130.42-1b, c), the design load combinations satisfy the corresponding SRP load combinations. However, under

*The draft revision to this table is to be included with the response to NRC Question 110.001. All information required for this response, however, is included with this response.

Service Load conditions (Table Q130.42-1a) certain design load combinations do not comply explicitly with the corresponding SRP combinations and these are considered below.

Design Combination 3 does not meet the requirements of SRP Combination 3. However, Design Combination 2 satisfies SRP Combination 2 and analysis shows that SRP Combination 2 controls over SRP Combination 3 for the concrete structures of 3.8.4. This follows because the seismic force in Combination 2 (1.9E), equal to 1.9 times the OBE acceleration times the component mass, is always greater than the design wind force in Combination 3 (1.7W), equal to 1.7 times the design wind force on the component. Hence, the design satisfies the requirements of SRP Combination 3.

The corresponding combinations with thermal loads, 2b and 3b, show only minor differences between the design combinations and the SRP combinations. Also, Combination 2, discussed above, controls with respect to both Combinations 2b and 3b. Hence, the design satisfies the requirements of the SRP combinations with thermal loads.

In the combinations with reduced gravity loads, 2b' and 3b', deviations between the design combinations and the SRP combinations exist. These combinations are intended to provide for the case when minimum downward loading is under consideration as in stability investigations. Analysis shows that smaller downward loads (more conservative loading) occurs with the design combinations as compared to the SRP combinations. This tends to equilibrate the overall effects of both sets of combinations. It is further noted that the design combinations comply precisely with the combinations of ACI 318-71 and are in accordance with the PSAR. The preceeding considerations justify the load combinations used with reduced gravity loads.

Steel Structures

The Seismic Category I steel structures outside containment are listed in 3.8.4.1. The steel components of these structures are considered herein. The load combinations applicable to these steel structures and components are listed in Table 3.8-16 of the FSAR. The table as submitted previously was used in the design when the applicable loadings did not involve pipe rupture loads. When pipe rupture loads are applicable, revised Table 3.8-16 lists the load combinations which were used.*

*Revised draft Table 3.8-16 is to be included in the response to NRC Question 110.001. All information required for this response, however, is included with this response.

A comparison of the load combinations in SRP 3.8.4 and those used in design is given in Tables Q130.42-2a, b, c. Service load conditions are covered in Table Q130.42-2a. Factored load conditions without pipe break and with pipe break are taken up in Tables Q130.42-2b, and 2c, respectively. The elastic working stress design method was used for service load conditions and factored load conditons without pipe break. The plastic design method was used for factored load conditions with pipe break. Symbols in the FSAR are used in the tables.

Review of the aforementioned tables shows that for Factored Load.Conditions with pipe break (Table Q130.42-2c), the design load combinations comply with the SRP load combinations. However, under Service Load Conditions (Table Q130.42-2a) and Factored Load Conditons without pipe break (Table Q130.42-2b), the design load combinations do not comply explicitly with the corresponding SRP combinations and these are considered below.

Thermal stresses due to T_o and R_o are not significant and these terms have not been explicitly listed in the load combinations. In addition, the live load term L has been explicitly listed only in Combination 1 and has been omitted from Combinations 2, 3, 4, and 5 which include seismic, wind, and tornado loads. This is justified on the basis that the live load magnitudes used in Combination 1 represent equipment laydown loads which are akin to construction loads and are short time in nature. Other "normal" live loads are much smaller in magnitude. It is noted that short time equipment lay-down loads are not normally included with seismic loads in current seismic design practice as evidenced in "Recommended Laterial Force Requirements and Commentary" by the Seismology Committee of the Strucural Engineers Association of California.

TABLE Q 130.42-1a

External Seismic Category I Concrete Structures –
Service Load Conditions
Pertinent Load Combinations –
Strength Design Method

NORMAL

Combination 1

SRP : $1.4\ D + 1.7\ L$
Table 3.8-15 : $1.4\ D + 1.7\ L$

Combination 2

SRP : $1.4\ D + 1.7\ L + 1.9\ E$
Table 3.8-15 : $1.4\ D + 1.7\ L + 1.9E + 1.7\ P_o$

Combination 3

SRP : $1.4\ D + 1.7\ L + 1.7\ W$
Table 3.8-15 : $1.1\ D + 1.3\ L + 1.3\ W + 1.3\ P_o$

WITH THERMAL LOADS

Combination 1b

SRP : $1.05\ D + 1.275\ L + 1.275\ T_o + 1.275\ R_o$
Table 3.8-15 : $1.4\ D + 1.7\ L + 1.4\ T_o + 1.4\ R_o + 1.7\ P_o$

Combination 2b

SRP : $1.05\ D + 1.275\ L + 1.425\ E + 1.275\ T_o + 1.275\ R_o$
Table 3.8-15 : $1.4\ D + 1.4\ L + 1.4\ E + 1.4\ T_o + 1.4\ R_o$
$+ 1.4\ P_o$

Combination 3b

SRP : $1.05\ D + 1.275\ L + 1.275\ W + 1.275\ T_o$
$+ 1.275\ R_o$
Table 3.8-15 : $1.1\ D + 1.3\ L + 1.3\ W + 1.1\ T_o + 1.1\ R_o$
$+ 1.3\ P_o$

TABLE Q 130.42-1a (Continued)

REDUCED GRAVITY LOADS

Combination 2b'

SRP : 1.2 D + 1.9 E
Table 3.8-15 : 0.9 D + 1.4 E

Combination 3b'

SRP : 1.2 D + 1.7 W
Table 3.8-15 : 0.9 D + 1.3 W

WITH SOIL AND HYDROSTATIC PRESSURES

SRP (ACI 318-71 Sections 9.3.4 & 9.3.5)
: 1.4 D + 1.7 L + 1.4 F + 1.7 Q

SRP (ACI 318-71 Sections 9.3.4 & 9.3.5)
: 0.9 D + 1.4 F + 1.7 Q

Table 3.8-15 Comb. U4
: 1.4 D + 1.7 L + 1.4 F + 1.7 Q + 1.4 T_o + 1.7 P_o

Table 3.8-15 Comb. U6
: 0.9 D + 1.4 F + 1.7 Q + 1.4 T_o

TABLE Q 130-42-1b

EXTERNAL SEISMIC CATEGORY I CONCRETE STRUCTURES - FACTORED LOAD CONDITIONS WITHOUT PIPE BREAK PERTINENT LOAD COMBINATIONS

Combination 4

SRP : $D + L + T_o + R_o + E'$
Table 3.8-15 : $D + L + T_o + R_o + E'$

Combination 5

SRP : $D + L + T_o + R_o + W'$
Table 3.8-15 : $D + L + T_o + R_o + W' + P'$
(Comb. U 11)

TABLE Q 130-42-1c

EXTERNAL SEISMIC CATEGORY I CONCRETE STRUCTURES - FACTORED LOAD CONDITIONS WITH PIPE BREAK PERTINENT LOAD COMBINATIONS

Combination 6

SRP : $D + L + T_a + R_a + 1.5\ P_a$
Table 3.8-15 : $D + L + T_a + R_a + 1.5\ P_a$
(Revised)

Combination 7

SRP : $D + L + T_a + R_a + 1.25\ P_a + R_r + 1.25\ E$
Table 3.8-15 : $D + L + T_a + R_a + 1.25\ P_a + R_r + 1.25\ E$
(Revised)

Combination 8

SRP : $D + L + T_a + R_a + P_a + R_r + E'$
Table 3.8-15 : $D + L + T_a + R_a + P_a + R_r + E'$
(Revised)

Response 130.42

TABLE Q 130.42-2a

External Seismic Category I ~~Concrete~~ Steel Structures -
Service Load Conditions
Pertinent Load Combinations -
Elastic Working Stress Design Method

Combination 1

SRP : D + L
Table 3.8-15 : D + L

Combination 2

SRP : D + L + E
Table 3.8-15 : D + E + P_o

Combination 3

SRP : D + L + W
Table 3.8-15 : D + W + P_o

TABLE Q 130.42-2b

External Seismic Category I Steel Structures -
Factored Load Conditions Without Pipe Break
Pertinent Load Combinations -
Elastic Working Stress Design Method

Combination 4

SRP : D + L + T_o + R_o + E'
Table 3.8-16 : D + E' + P'
(Comb. U 13)

Combination 5

SRP : D + L + T_o + R_o + W'
Table 3.8-16 : D + W' + P'
(Comb. U 14)

Response 130.42

TABLE Q 130.42-2c

External Seismic Category I Steel Structures –
Factored Load Conditions With Pipe Break
Pertinent Load Combinations –
Plastic Design Method

Combination 6

SRP : $D + L + T_a + R_a + 1.5\ P_a$
Table 3.8-16 (Revised) : $D + L + T_a + R_a + 1.5\ P_a$

Combination 7

SRP : $D + L + T_a + R_a + 1.25\ P_a + R_r + 1.25\ E$
Table 3.8-16 (Revised) : $D + L + T_a + R_a + 1.25\ P_a + R_r + 1.25\ E$

Combination 8

SRP : $D + L + T_a + R_a + P_a + R_r + E'$
Table 3.8-16 (Revised) : $D + L + T_a + R_a + P_a + R_r + E'$

~~WPPSS~~

~~See Q. 110.01 for revised Tables 3.8-15 and 3.8-16.~~

Q. 130.043
(3.8.5)

Indicate whether you used Section III, Division 2 of the ASME Boiler and Pressure Vessel Code in your analysis and design of the reactor building foundation mat. If not, provide justification for this deviation and assess the significance of any deviation from this code with respect to the structural adequacy of the foundation mat of the WNP-2 reactor building.

Response:

The WNP-2 containment is a steel containment with its bottom continuous through an inverted dome. Consequently, foundation design requirements of Regulatory Guide 1.70, Paragraph 3.8.2.1, Subparagraph 1.a. are applicable with requirements as follows:

a. The method by which the inverted dome and its supports are anchored to the concrete foundation should be described. This is done in 3.8.2 of the FSAR.

b. The concrete foundation should be in accordance with 3.8.5.

In line with the above, 3.8.5 of the Standard Review Plan provides the applicable design requirements for the reactor building foundation mat. Compliance with ACI-318-71, "Building Code Requirements for Reinforced Concrete" is required. Section III, Division 2 of the ASME Boiler and Pressure Vessel Code is not applicable. As noted in the FSAR, 3.8.5.2 and Table 3.8-9 the WNP-2 reactor building foundation mat design has been based on ACI-318-71.

Q. 130.044
(3.8.5)

Indicate whether there is any uplifting (i.e., tilting from the horizontal) predicted for the foundation mats of all Seismic Category I structures. If so, identify these structures and indicate the amount of the estimated uplifting.

Response:

Please refer to revised 3.8.5.5 for the information requested.*

*Draft FSAR page changes attached.

mat foundation by perimeter foundation walls, interior shear walls and columns through shear keys and shear friction reinforcement. The mat beam strips are computer analyzed by FLXMAT, a proprietary computer program developed by Burns and Roe, Inc., for the analysis of flexible mat foundations on elastic foundations and is discussed in 3.12.

3.8.5.4.7 Makeup Water Pumphouse

The mat foundation at the bottom of the pump pit and the continuous wall footings are conventionally analyzed. The mat foundation is analyzed as a two-way slab supported on four (4) sides by the pump pit walls.

3.8.5.5 Structural Acceptance Criteria

Foundations are designed in compliance with ACI 318-71 and satisfy the strength and the serviceability requirements specified therein. The structural acceptance criteria for reinforced concrete described in 3.8.4.5.1 is applicable to the foundation designs.

The factors of safety against overturning and sliding for safety-related structures are as follows:

Safety-Related Structure	Safety Factors Overturning	Sliding
Reactor building	1.50	1.80
Radwaste and Control Building	3.13	2.29
Turbine Generator Building	6.80	2.70
Standby Service Water Pumphouses	1.60	3.55

Uplifting of foundation mats occurs for the Seismic Category I structures listed below. Maximum uplift occurs as a result of the combined effects of horizontal and upward safe shutdown earthquakes (SSE). The maximum uplift conditions are described below.

a. Reactor Building - Under maximum uplift, 48 percent of the mat maintains bearing contact with the soil. Maximum upward deflection is 1.1 inches.

b. Turbine-Generator Building - Under maximum uplift, 89 percent of the mat maintains bearing contact

with the soil. Maximum upward deflection is 0.10 inches.

c. Radwaste and Control Building - Under maximum uplift, 84 percent of the mat maintains bearing contact with the soil. Maximum upward deflection is 0.14 inches.

RESPONSES TO

REACTOR SYSTEMS BRANCH (RSB I)

QUESTIONS 211.002 - 211.048

Q. 211.002
(5.2.5)
(7.6.1)
(9.3.3)

Discuss the sump geometry, the accuracy of the leakage flow rate measurements, the monitoring interval and any other information required to demonstrate a sensitivity of one gallon per minute (gpm) per hour for the floor drain sump level monitoring systems.

Response:

The floor drain sump system is a gravity flow feed from the drywell floor drain into the reactor building floor drain sump. The floor drain sump geometry is such that the gravity feed is connected to an overflow standpipe in the center of the sump. Approximately 50 gallons of liquid is required to fill a completely dry sump to the overflow gravity feed inlet. Thus, the required sensitivity of 1 gpm per hour can be monitored. A flow transmitter continually measures flow from the drywell floor drain and records this information in the control room. The recorder actuates a control room alarm if flow exceeds a pre-established limit.

The system is designed to detect a flow as little as 0.5 gpm with a total system accuracy of 1.8%.

211.003
(5.2.5)

The design of the WNP-2 facility routes drainage of both "hot" and "cold" reactor coolant leakage into the drywell equipment drain sump. However, relatively hot sources of leakage water (e.g., the reactor vessel head flange, the vent drain and valve packings) may flash into steam which then must be condensed before it can be drained into the sump. Accordingly, indicate what assurance there is that this steam from relatively hot sources will be condensed so that monitoring of this leakage may be performed in the drywell sump to detect these "hot" leaks. Since leakage from those sources which are relatively cool is drained into the floor drain system, this system should be tested periodically for blocked lines. Accordingly, discuss the surveillance program you propose for the WNP-2 facility to detect blockage of the floor drain system.

Response:

WNP-2 routes "hot" reactor coolant leakage to the drywell equipment drain sump through leakoff lines to a main header. The header discharge is routed to the equipment drain condenser where "hot" leakage is cooled and condensed. Discharge from the condenser to the sump is through a quencher. This arrangement assures that any "hot" leakage which could flash into steam will be cooled and condensed before entering the open sump.

The floor drain sump drain line from the containment to the reactor building sump will be periodically back flushed to show that the drain line is not blocked.

Q. 211.004
(5.2.1)
(7.6.2)

In conformance with the staff's position in Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", May 1973, you state in Section 7.6.2.4 of the FSAR that the radioactivity monitoring channels are qualified for operation following a Safe Shutdown Earthquake (SSE). Confirm that all of the other leakage detection methods and/or systems will function properly following an Operating Basis Earthquake (OBE). These other leakage detection systems include the drywell equipment sump and the floor drain sump, the sump coolers, and the associated instrumentation and piping.

Response:

The WNP-2 drywell floor drain and equipment drain sumps, piping to the sumps, and the equipment drain cooler are seismically supported such that they will continue to pass leakage flow following an OBE.

The flow monitoring instrumentation is not qualified to withstand an OBE. WNP-2 will initiate a design change to qualify the existing flow monitoring instrumentation to assure its accuracy following an OBE.

Q. 211.005
(5.2.5)

Provide a list of the normal and the maximum anticipated leakage rates through the reactor coolant pressure boundary (RCPB), including the concentrations of radioactivity in this leakage flow, from both identified and unidentified sources (e.g., the control rod drive flanges and the vent cooler drains) which are routed into the drain sumps.

Response:

The normal and maximum anticipated leakage rates from the reactor coolant pressure boundary within the drywell during normal reactor operation is between 0.1 and 0.5 gpm for unidentified leakage and 2 to 5 gpm for identified leakage. These leakage rates are measured at the sumps and are based on operating plant experience.

The activity concentrations in this leakage are expected to be the same as those in the reactor water. The concentrations are given in Chapter 11, Tables 11.1-2, "Halogen Radioisotopes in Reactor Water"; 11.1-3, "Other Fission Product Radioisotopes in Reactor"; 11.1-4, "Coolant Activation Products in Reactor Water and Steam"; and 11.1-5, "Noncoolant Activation Products in Reactor Water".

Q. 211.006
(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 response to this question is pending on the sensitivity curves which are to be prepared by the equipment vendor, Kaman Sciences Corporation. These curves have not yet been received by WPPSS. The submittal date to the NRC of the final response to this question will depend on the receipt of the aforementioned curves.

Q. 211.007
(7.6.2)

Expand your discussion in Section 7.6.2.4 of the FSAR regarding the operability verification and the calibration of the leakage detection system which will be accomplished by comparing the results of diverse monitoring methods. For example, if a radioactivity monitoring system is checked against the sump level and the flow monitoring system, indicate how the latter system is determined to have acceptable accuracy. Confirm that calibration and operability tests will: (1) be performed periodically during plant operation; and (2) be in compliance with the requirements of IEEE Standard 279-1971.

Response:

Comparisons between the drywell floor drain flow monitoring system and radioactivity monitoring systems will be made during plant operation. The sensitivity of the monitors are within the guidelines of Regulatory Guide 1.45 (May, 1973) whereas the accuracy of the measurements are influenced by the fission product and corrosion-coolant activation product concentrations in the primary coolant steam and water phases. Operational experience is required to establish the acceptable accuracy for the system at the various operating conditions.

That equipment which provides inputs to safety-related systems are designed to IEEE Standard 279-1971 as stated in 7.6.2.4.2.3.a and as such will have operability and calibration tests performed as stated in 7.6.2.4.2.3.c.

Miscellaneous revisions to 7.6.2.4 and 5.2.5 have been made in light of this question.*

*See attached draft page changes.

5.2.5 DETECTION OF LEAKAGE THROUGH REACTOR COOLANT PRESSURE BOUNDARY

5.2.5.1 Leakage Detection Methods

The nuclear boiler leak detection system consists of temperature, pressure, and flow sensors with associated instrumentation and alarms. This system detects, annunciates, and isolates (in certain cases) leakages in the following systems:

a. Main Steam Lines

b. Reactor Water Cleanup (RWCU) system

c. Residual Heat Removal (RHR) system

d. Reactor Core Isolation Cooling (RCIC) system

e. Feedwater system

f. High Pressure Core Spray

g. Low Pressure Core Spray

h. Coolant system within the Primary Containment

Isolation and/or alarm of affected systems and the detection methods used are summarized in Table 5.2-9.

Small leaks (5 gpm and less) are detected by temperature and pressure changes and drain sump pump activities. Large leaks are also detected by changes in reactor water level and changes in flow rates in process lines.

The 5 gpm leakage rate is a proposed limit on unidentified leakage. The leak detection system is fully capable of monitoring flowrates of one gpm and is, thus, in compliance with Paragraph C.2 of Regulatory Guide 1.45.

5.2.5.1.1 Detection of Abnormal Leakage Within the Primary Containment

Leaks within the drywell are detected by monitoring for abnormally high pressure and temperature within the drywell, high fillup rates of equipment and floor drain sumps, excessive temperature difference between the inlet and outlet cooling water for the drywell coolers, ~~increased flow rate of the cooler condensate~~, a decrease in the reactor vessel water level, and high levels of fission products in the

~~c. Drywell Cooler Drain~~

~~Condensate from the drywell coolers is routed to the floor drain sump and is monitored by use of a flow transmitter mounted locally while having indicating and alarm instrumentation in the control room. An adjustable alarm is set to annunciate when the condensate flow rate approaches the technical specification limit.~~

c ~~f~~. Drywell Pressure Measurement

The drywell is at a slightly positive pressure during reactor operation. The pressure fluctuates slightly as a result of barometric pressure changes and outleakage. A pressure rise above the normally indicated values will indicate the presence of a leak within the drywell.

d ~~g~~. Drywell Temperature Measurement

The drywell cooling system circulates the drywell atmosphere through heat exchangers (air coolers) to maintain the drywell at its designed operating temperature and also provides cooling water to the air coolers. An increase in drywell atmosphere temperature would increase the temperature rise in the service water passing through the coils of the air coolers. Thus, an increase in the service water temperature difference between inlet and outlet to the air coolers will indicate the presence of reactor coolant or steam leakage. Also, a drywell ambient temperature rise will indicate the presence of reactor coolant or steam leakage. A temperature rise in the drywell is detected by monitoring the drywell temperature at various elevations, inlet and outlet air to the coolers, and the closed cooling water temperature increase between inlet and outlet to the coolers.

e ~~h~~. Drywell Air Sampling

The drywell air sampling system is used to supplement the temperature, pressure, and flow variation method described previously to detect leaks in the nuclear system process barrier. The system continuously monitors the drywell atmosphere for airborne radioactivity. The sample is drawn from the drywell. A sudden

increase of activity, which may be attributed to steam or reactor water leakage, is annunciated in the control room. (Refer to Containment Atmospheric Monitoring System, 7.6.1.13.)

Table 5.2-9 summarizes the actions taken by each leakage detection function. The table shows that those systems which detect gross leakage initiate immediate automatic isolation. The systems which are capable of detecting small leaks initiate an alarm in the control room. The operator can manually isolate the violated system or take other appropriate action.

f ~~g.~~ Reactor Vessel Head Closure

The reactor vessel head closure is provided with double seals with a leak off connection between seals that is piped to the equipment drain sump. Leakage through the first seal is annunciated in the control room. When pressure between the seals increases, an alarm in the control room is actuated. The second seal then operates to contain the vessel pressure.

g ~~h.~~ Reactor Water Recirculation Pump Seal

Reactor water recirculation pump seal leaks are detected by monitoring the drain line. Leakage, indicated by high flow rate, alarms in the control room. Leakage is piped to the equipment drain tank.

h ~~i.~~ Safety/Relief Valves

Temperature sensors connected to a multipoint recorder are provided to detect safety/relief valve leakage during reactor operation. Safety/relief valve temperature elements are mounted, using a thermowell, in the safety/relief valve discharge piping several feet from the valve body. Temperature rise above ambient is annunciated in the main control room. See the nuclear boiler system piping and instrumentation diagram, Figure 5.1-3a, b, c.

i X. Valve Packing Leakage

Valve stem packing leaks of power-operated valves in the nuclear boiler system, reactor water cleanup system, high pressure core spray, low pressure core spray, reactor core isolation cooling system, residual heat removal system, and recirculation system are detected by monitoring packing leakoff for high temperature and are annunciated by an alarm in the control room.

5.2.5.3 Indication in Control Room

Leak detection methods are discussed in 5.2.5.1. Details of the leakage detection system indications are included in 7.6.1.4.3.

5.2.5.4 Limits for Reactor Coolant Leakage

5.2.5.4.1 Total Leakage Rate

The total leakage rate consists of all leakage, identified and unidentified, that flows to the drywell floor drain and equipment drain sumps. The criterion for establishing the total leakage rate limit is based on the makeup capability of the RCIC system. The total leakage rate limit is established at 30 gpm, 25 gpm identified and 5 gpm unidentified.

The total leakage rate limit is also set low enough to prevent overflow of the drywell sumps. The equipment sump and the floor drain sump, which collect all leakage, are each drained by two 50-gpm pumps.

5.2.5.4.2 Normally Expected Leakage Rate

The pump packing glands, valve stems, and other seals in systems that are part of the reactor coolant pressure boundary and from which normal design leakage is expected are provided with drains or auxiliary sealing systems. Nuclear system valves and pumps inside the drywell are equipped with double seals. Leakage from the primary recirculation pump seals is piped to the equipment drain sump as shown in Figure 5.4-2b. Leakage in the discharge lines from the main steam safety/relief valves is monitored by temperature sensors that transmit a signal to the control room. Any temperature increase above the drywell ambient temperature detected by these sensors indicates valve leakage. Leakage from the reactor vessel head flange is also monitored (see 7.6.1).

7.6.2.4.2.1.2 Regulatory Guide 1.45 (5/73)

The leakage to the primary reactor containment from identified sources such as valve stem packing, recirculation pump seal, fuel storage pool, head seal, etc. is separated so that flow rates are monitored separately from unidentified leakage and total flow rate can be established and monitored. The leakage from the main steam line safety/relief valves is identified leakage because of the location of the sensors which detect this leakage, but the leakage is not completely separated from unidentified sources. Separation of this leakage is not required since any leak from the main steam line safety/relief valves would not be from a crack or break in the line so there would be no identified leakage from the S/R valve lines during plant operation which necessitates separation from unidentified leakage. The leakage to the reactor containment from unidentified sources is collected and this flow rate is monitored with an accuracy of better than one gallon per minute.

The following required detection methods are used to monitor unidentified leakage:

a. Sump level and flow monitoring

b. Airborne particulate radioactivity monitoring

~~c. Air cooler condensate flow rate monitoring, and~~

c ~~d~~. Airborne gaseous radioactivity monitoring.

Provisions are made to monitor systems connected to the RCPB for signs of intersystem leakage, including radioactivity monitoring of process fluids (process rad system) and reactor vessel water level monitoring (NSSS).

The sensitivity and response time of each system for detection of unidentified leakage is one gallon per minute in less than one hour, except for the airborne particulate radioactivity and airborne gaseous activity monitoring channels, which have sensitivities of 10^{-9} μCi/cm3 and 10^{-6} μCi/cm3 respectively, which are the sensitivities suggested for these channels by Regulatory Guide 1.45.

The leakage detection system is qualified for operation following an OBE. The particulate radioactivity monitoring channel is qualified for operation following an SSE.

Indicators and alarms for each leakage detection subsystem are provided in the main control room. At the site, procedures for converting various indications e.g., temp, WT, and pressure, to a flow rate measurement will be provided by means of conversion curves wherever meaningful.

The leakage detection system is equipped ~~with provisions~~ to readily verify operability and calibration by means of correlation with diverse monitoring methods.

Major components within the drywell that by nature of their design are sources of leakage (e.g., sump seals, valve stem packing), are contained and piped to an equipment drain sump and thereby identified.

Equipment associated with systems within the drywell (e.g., vessels, piping, fittings) share a common free volume, therefore, their leakage detection systems are common. Steam or water leaks from such equipment are collected ultimately in an area drain sump.

Each of the sumps are protected against overflowing leaks from one source masking those from another.

The area drains collecting system is designed to detect leakage in excess of 1 gpm within 1 hour.

As added back-up to the unidentified leakage drain system, the main steam lines within the steam tunnel inside the containment are monitored by temperature detectors within the tunnel.

In summary, the leak detection system meets the requirements of Regulatory Guide 1.45 both in the variables measured and in the sensitivity of the measurements.

7.6.2.4.2.1.3 Regulatory Guide 1.47 (5/73)

The leak detection system annunciates all bypass conditions.

7.6.2.4.2.1.4 Regulatory Guide 1.53 (6/73)

The leak detection system complies with this guide. Discussion is provided in 7.3.2.2.2.1.6 under Regulatory Guide 1.53.

Q. 211.008
(5.2.2)
(7.6.1)

In Section 7.6.1.4 of the FSAR, you state that major components within the drywell which are sources of leakage by nature of their design (e.g., the sump seals, the valve stem packing, and the equipment warming drains), are enclosed and the leakage is piped to an equipment drain sump and identified there. Indicate what you mean by the term "sump seals" (i.e., did you intend to state pump seals). Discuss what monitors are available to the operator to permit him to identify the source of leakage; i.e., whether the leakage is from the "sump" (or pump) seals or the equipment warming drains or from any other component leakage sources which drain into the drywell equipment drain tank. Indicate whether there are any sumps within the drywell which must be filled before the sump drain flow is routed to the equipment drain tank.

Response:

The term "sump seals" is an error and should read "pump seals".

Sources of leakage which are enclosed and piped to the Drywell Equipment Drain Sump are monitored by the following means to provide the operator with information to permit him to identify the source of leakage.

(1) Recirculation pump seals are provided with flow switches and control room alarms to identify excessive seal leakage.

(2) The area between the vessel head seals is monitored with a pressure switch and a control room alarm to indicate inner seal leakage.

(3) Remote operated valves within the Drywell are provided with seal leakoff lines. These lines connect between the valve stem packing are equipped with thermocouples to detect leakage through the inner packing and alarm in the control room.

All other sources of leakage which are piped to the drywell equipment drain tank are not considered as leakage paths. These lines are isolated by double block valves, which are opened only when needed.

Both the drywell equipment drain and floor drain sumps must fill to a specified level before reaching an overflow which routes liquid outside the drywell. See the response also to Question 211.002.

Q. 211.009
(5.2.5)
(7.6.2)

In conformance with the staff's positions in Regulatory Guide 1.45, you state that the positions will be made to monitor systems connected to the reactor coolant pressure boundary for signs of intersystem leakage. Provide a detailed discussion of these provisions, including an identification of all potential intersystem leakage paths; e.g., the leakage from the primary coolant system to the residual heat removal (RHR) system and the emergency core cooling system (ECCS) injection line. Identify the instrumentation used in each path which will provide positive indication of intersystem leakage in the affected system.

Response:

Below is a listing of equipment which alert the operator to intersystem leakage from the reactor coolant pressure boundary into a low pressure system.

Each low pressure system process line is provided with an overpressure sensing pressure switch. The pressure switch activates an alarm in the control room to alert the operator to a degrading situation where a leaking shutoff valve may result in system pressure exceeding design limits.

The following is a list of process line, line design limits, associated overpressure sensor, and sensor set-point activating an alarm.

Intersystem Leakage Path	Design Pressure	Sensor ID No.	Sensor Setpoint
RHR/Recirc Suction	220 psig	PS E12-N018	190 psig
RHR/Recirc Discharge	500 psig	PIS E12-N022A,B	400 psig
RHR Head Spray	500 psig	PIS E12-N022B	400 psig
LPCS Injection	550 psig	PS E21-N005	450 psig
LPCI Injection	500 psig	PIS E12-N022A,B,C	400 psig

Q. 211.010
(5.2.5)

Operating experience at some boiling water reactors (BWRs) has shown that the high pressure coolant injection system (HPCI) and the reactor core isolation cooling system (RCIC) have been rendered inoperable due to inadvertent leak detection isolations which were caused by a high differential temperature signal from the equipment room area. These isolations occurred when there was a relatively sharp drop in the outside temperature. In 5.4.6.1.1.1 and Table 5.2-9 of the FSAR you indicate that the WNP-2 facility also has this type of isolation for the RCIC system and the steam side of the RHR system. Provide a discussion of the modifications that have been, or will be, made to prevent inadvertent isolations of this type which effect the availability and reliability of the RCIC and the RHR systems. Additionally, indicate the trip settings you propose for isolation of the WNP-2 RHR and RCIC systems due to high area temperature in terms of degree Fahrenheit above the ambient temperature. Discuss the method you propose to avoid this problem. Show that the differential temperature setting could not be set too low, thereby causing an inadvertent isolation when the RCIC and the RHR systems are needed.

Response:

The RHR and RCIC pump rooms are located approximately 20 feet below grade level and would not be directly affected by a sharp drop in the outside air temperature. These rooms are ventilated by the reactor building ventilation system which contains a freeze protection device that will annunciate in the main control room if 40°F air existed downstream of the heating coil.

The high differential temperature setting for both RHR and RCIC pump rooms is 70°F. The temperature elements are located at the face of the supply and exhause ductwork in each room. The maximum room temperature rise could be 58°F, in one of the three RHR pump rooms, with RHR pump operating. It is, therefore, very unlikely that an inadvertent isolation of the RHR or RCIC systems could occur.

The high temperature trip setting for RHR and RCIC pump rooms is 150°F which is 15°F higher than the maximum calculated room temperature.

If any of the trip settings are too low, the consequences have no safety significance, because RCIC and the steam supply to the RHR heat exchanger are not required safety systems. The trip setting will be adjusted if necessary to reduce any operational inconveniences.

Q. 211.011
(5.5.2)
(Appendix C-2)

You deviate from the Staff's positions in Regulatory Guide 1.29, Revision 3, "Seismic Design Classification", September, 1978, by not designing the component cooling water portions of the reactor recirculation pumps to seismic Category I criteria. The basis you state in the FSAR for this deviation is that these pumps do not perform a safety function. Provide additional justification for this position and show that a loss of component cooling water to the recirculation pumps would not lead to unacceptable consequences.

Response:

Seismic Category I applies to the pressure boundary at the pump. The pump is not required to be operational during safe reactor shutdown and, therefore, the consequences of loss of cooling water to the recirculation pumps are acceptable.

There is a requirement for coastdown capability for the hypothesized loss-of-coolant accident. The equipment has demonstrated adequate coastdown time in the past following loss of cooling water.

Q. 211.012
(4.6)
(5.4.6)
(5.4.7)

Describe the provisions incorporated into the WNP-2 facility to protect the RCIC and the RHR systems from cold weather and from dust storms and to assure satisfactory operational performance under any adverse meteorological conditions. In this discussion, include consideration of the standby liquid control system and the control rod drive (CRD) hydraulic system and any other sources of water for these systems (e.g., the condensate storage tank and the standby service water).

Response:

The RCIC system taken suction from the condensate storage tanks during normal modes of operation. The condensate storage tanks are provided with heaters to maintain water temperature above 40°F at all times. All above ground piping that contains water is heat traced to prevent freezing. Since the CST is a covered tank, the water supply is not affected by dust storms. To provide a Category I source of cooling water for the RCIC system, an alternate path or cooling water can be valved in from the suppression pool, which is inside the reactor building and protected from cold weather and dust storms.

The control rod drive hydraulic system normally takes suction from the main condensate system, downstream of the condensate demineralizers. All the piping is located within the Turbine Building or Reactor Building. The secondary source of water is the condensate storage tank if the main condensate system is not available. Both sources of water are protected from cold weather and dust storms.

The standby liquid control system, which is filled with sodium pentaborate, is provided with tank heaters and heat tracing to prevent solidification. The entire system is located within the Reactor Building, so it is unaffected by cold weather or dust storms.

The RHR system takes suction from either the recirculation piping or the suppression pool. All the piping is within the Reactor Building.

The RHR heat exchangers dissipate their heat to the standby service water system. All SW piping and components are either below the frost line, within the heated pumphouse, or, in the case of the spray rings, kept drained by the return header drain valve when not in operation. The SW pump suction is 26 feet below water level, so any dust collecting on the spray pond surface will not affect pump operation. A sand trap is located in front of each SW pumphouse. Any equipment that could be affected by dust storms is either provided as sealed units, located in dust-proof cabinets, or protected by dust-proof coatings. (See the response to Question 010.016.)

Q. 211.013
(3.5.1)

In evaluating the potential for missiles due to failures of pressurized components, you state that thermowells and sample probes have been assessed against criteria discussed in Section 3.5.1.1.2 of the FSAR. Indicate which specific criterion and basis has been considered in determining that the thermowells and sample probes are not credible missiles. Provide justification for omitting other pressurized components such as blank flange assemblies and pressurized vessels or bottles (e.g., the safety/relief valve air accumulators and the nitrogen accumulator tanks) from your assessment of potential missiles.

Response:

Thermowells and sample probes are investigated for their potential of becoming postulated missiles which could adversely affect the shutdown capability of the plant.

Thermowells that are in systems where the pressure or temperature is 275 psig, 200°F or greater, respectively, are analyzed for their potential of becoming missiles.

The connections between the thermowells and piping systems have been analyzed and determined to have ample factors of safety against structural failure. For conservatism, however, thermowells are postulated as missiles, and it is found that even if all systems and components in the path of the missile are assumed lost, the plant can still be brought to a safe (cool) shutdown.

In the WNP-2 facility sample probes are lines that are one inch or less in diameter, and pipe breaks are not postulated in accordance with SRP 3.6.26 (MEB 3-1).

Blank flange assemblies, pressurized vessels and bottles, are investigated as to their potential of becoming missiles with the procedure outlined in 3.5.1.1.2 of the FSAR.

Q. 211.014
(3.5.1)

Discuss the potential for missiles inside the containment due to falling objects (e.g., electrical hoists or any unrestrained equipment) for the following events: (1) routine maintenance; (2) reactor operation; and (3) a postulated loss-of-coolant accident (LOCA).

Response:

A discussion of missiles inside containment due to falling objects was provided in Amendment 3 (3.5.1.7.4) in response to NRC Question 212.001. As stated in 3.5.1.2.4, the only components inside containment which are not supported to satisfy Seismic Category I requirements are the monorail hoists which are chained in place while the reactor is operating.

During routine maintenance, appropriate precautions will be taken to minimize the vulnerability of systems required to maintain the plant in a safe shutdown condition, while loads are being moved about. Such precautions could consist of:

1) Routing and providing temporary storage of loads to minimize travel or positioning of loads over vulnerable components of essential systems.

2) Providing temporary barriers for essential equipment.

3) Providing redundant slings or chains used in carrying loads.

4) Tying off potential falling objects which are being temporarily stored. Conditions existing at the time maintenance is performed will dictate whether precautions such as those described above need be taken, and the extent of such precautions.

Pipe whip restraints for high energy systems are provided to restrain the motion of ruptured pipe, and prevent further damage which could result in missile formation, as described in 3.6.2. Design for jet impingement effects, as described in 3.6.2, has not included an assessment of

missile formation and consequences, due to impact by postulated jets, for the same reasons provided in 3.5.1.2.5 as to why analyses of secondary missiles caused by primary missiles are not performed. However, the physical separation and redundancy of safe shutdown systems, as described in 3.5.1.2.2, provides assurance that if missiles or falling objects were generated by a LOCA, that the ability to bring the plant to a safe shutdown would be maintained.

Q. 211.015
(3.5.1)

With regard to missiles generated by a postulated failure of a rotating component, show by analysis that the impeller fragments resulting from an overspeed condition in the recirculation pump during a postulated LOCA, will not penetrate the pump case. Provide a study demonstrating that the probability for significant damage to safety-related components or systems inside containment resulting from impeller missiles which might be ejected out the open end of the broken pipe, is acceptably low. If you reference a similar study on another docket, demonstrate the appropriateness of referencing this study for the WNP-2 facility.

Response:

Reference 3.5-4 has been revised to refer to Revision 2 of the General Electric Company Report, "Analysis of Recirculation Pump Under Accident Conditions". This report concludes that pump overspeed due to postulated pipe breaks is highly improbable, and that in the unlikely event of impeller failure the missile fragments lack sufficient energy to penetrate the pump case. The effects of impeller missile fragments ejected from openings in ruptured recirculation system piping are not evaluated because they would not be more severe than the assumed consequences of jet impingement from the same breaks. Furthermore, except for a circumferential break in the first straight run of piping at the discharge side of the recirculation pump, contact with the pipe due to directional changes by the impeller fragment prior to ejection from the broken pipe would result in substantial energy loss. For these reasons, recirculation pump missile fragments are not included as postulated missiles inside containment. See revised Section 3.5.1.2.3.4.*

*Draft FSAR page change attached.

3.5.1.2.3.1 Valves

Valves have been investigated as discussed in 3.5.1.1.3.1, and it was found that the ability to shut down the plant is not affected.

3.5.1.2.3.2 Thermowells and Sample Probes

Thermowells and sample probes have been examined as discussed in 3.5.1.1.3.2, and it was found that the ability to shutdown the plant is not affected.

3.5.1.2.3.3 Bolts

Bolt failures are unlikely to occur for reasons discussed in 3.5.1.1.3.3, and it was found that the ability to shut down the plant is not affected.

3.5.1.2.3.4 High Speed Rotating Equipment

a. Recirculation Pump and Motor

The most substantial piece of NSSS rotating machinery is the recirculation pump and motor. This potential missile source is covered in detail in reference 3.5-4.

It is concluded in reference 3.5-4 that destructive pump ~~overspeed can result in certain types of missiles. A careful examination of shaft and coupling failures shows that the fragments will not result in damage to the containment or to vital equipment. Therefore, these fragments are not postulated as missiles. The following are postulated missiles from rotating equipment:~~

~~a. Low Energy Missiles (Kinetic energy less than 1,000 ft-lbs)~~

~~Low energy level missiles may be created at 300% of rated motor speed, through failure of the end structure of the rotor. The structure consists of the retaining ring, the end ring, and the fans. Missiles potentially generated in this manner will strike the overhanging ends of the stator coils, the stator coil bracing, support structures, and two walls of one-half inch thick steel plate. Due to the ability of these structures to absorb energy, it is~~

Replace with insert $\boxed{A}$

Insert A:

overspeed is highly improbable. If it occurred it could result in failure of certain pump and motor components having the potential to become missiles. A careful examination of the pump and motor structure shows that rotor or shaft failure will not result in ejection of motor-generated missiles, and impeller missiles cannot penetrate the pump case. Reference 3.5-4 concludes that in the unlikely event of impeller failure resulting in ejection of missiles through ruptured pipe, penetration of containment by missile fragments is highly improbable. Evaluation of the effects on safety-related systems of impeller fragments which might be ejected from openings in ruptured pipe is not evaluated because of the extreme improbability of this event, and because the effects would not be more severe than the assumed consequences of jet impingement due to pipe breaks inside containment is discussed in 3.6.

~~concluded that missiles would not escape this structure. It is at this point frictional forces would tend to bring the overspeed sequence to a stop.~~

DELETE

~~b. Medium Energy Missiles (Kinetic energy less than 20,000 ft-lbs)~~

~~In the postulated event that the body of the rotor were to burst, medium energy missiles could be created. The likelihood that these missiles would escape the motor is considered less than the likelihood of escape for the low-energy missiles described above, due to the additional amount of material constraining missile escape, such as the stator coil, field coils, and stator frame directly adjacent to the rotor.~~

~~c. The Motor as a Potential Missile~~

~~Since bolting is capable of carrying greater torque loads than the pump shaft, pump bolt failure is precluded. Since pump shaft failure decouples the rotor for the overspeed driving blowdown force, only those cases with peak torques less than that required to fail the pump shaft (five times rated) will have the capability to drive the motor to overspeed. When missile generation probabilities are considered along with a discussion of the actual load bearing capabilities of the system, it is evident that these considerations support the conclusion that it is unrealistic that the motor would become a missile.~~

b. ~~d.~~ Fans as Potential Missiles

The fans inside primary containment are designed such that the casing will restrain any possible missile. Therefore, fans and parts thereof are not considered as possible sources of missiles.

3.5.4 REFERENCES

3.5-1 Gwaltney, R. C., "Missile Generation and Protection in Light Water Cooled Power Reactor Plant" Oak Ridge National Laboratory.

3.5-2 Amerikan, A., "Design of Protective Structures", Bureau of Yards and Docks" No. NAVDOCKS P-51, 1950.

3.5-3 Cottrell, Wm. B., Savolainen, A. W., U.S. Reactor Containment Technology, Oak Ridge National Laboratory and Bechtel Corporation, August 1965.

3.5-4 ~~Letter, E. A. Hughes (GE) to R. C. Young (USNRC), January 18, 1977, GE Recirculation Pump Potential Overspeed.~~

3.5-5 Regulatory Guide 1.115, Rev. 0, "Protection Against Low Trajectory Turbine Missiles".

3.5-6 Westinghouse Report Covering the Effects of a Turbine Accelerating to Destructive Overspeed, Revised 1969, 1975.

3.5-7 S. H. Bush, Probability of Damage to Nuclear Components Due to Turbine Failure, U.S. Nuclear Regulatory Commission, Washington, D.C., November 1972.

3.5-8 Bates, Swanson, "Tornado Design Considerations for Nuclear Power Plants", American Nuclear Society, November 1967.

3.5-9 Biggs, J. M., Introduction to Structural Dynamics, McGraw Hill, New York, 1964, Chapter 2.

3.5-10 Air Force Design Manual, "Principles and Practices for Design of Structures', AFSNC-TDR-62-138, December 1962, Chapter 7.

3.5-4 Langley, Donald K., Robare, David J., Analysis of Recirculation Pump under Accident Conditions, Rev. 2, General Electric Company, March 30, 1979. Transmittal letter, D.J. Robare of G.E. to D.B. Vassallo of NRC, March 30, 1979.

Q. 211.016
(3.5.1)

Based on our review of the design integrity of nuclear power plant piping systems, we have noted several failures of safety valve headers which caused the valves to become missiles. (Refer to NUREG-0307, "Review and Assessment of Research Relevant to Design Aspects of Nuclear Power Plant Piping Systems", July 1977.) Since you address only the credibility of valve bonnets and stems as potential missiles, provide justification in Section 3.5.1.2 of the FSAR for concluding that the safety valve header and valve cannot be considered to be credible missiles. Your statement in Section 3.5.1.1.3.1 of the FSAR that bonnet ejection is highly improbable and that bonnets are not considered to be credible missiles, is not supported. Show that should a large valve component become a missile, containment penetration would not occur. Discuss the provisions incorporated into the WNP-2 facility (e.g., equipment separation and redundancy) to preclude damage to the systems necessary to achieve and maintain a safe plant shutdown.

Response:

Safety evaluations based upon a redundancy study of systems, equipment and components, which had been physically arranged as of mid-1976, are currently being updated to reflect the final and completed arrangements of these items. The full detailed response to NRC Question 211.016 awaits the results of this completed study and will be presented in an amendment to the FSAR.

The primary provisions incorporated into the design of the WNP-2 facility to preclude damage to the systems necessary to achieve and maintain a safe plant shutdown are separation and redundancy. These provisions provide that in the event of an accident plus an additional active component, failure where a system required for safe (cool) shutdown is rendered unavailable, enough systems are left available to bring the plant to a safe (cool) shutdown without allowing any offsite radiological consequences. This redundancy and separation are obtained by the deliberate routing of systems, by the presence of structural floors, walls, structural stell members and adjacent equipment which serve as barriers. It is also obtained by discreet orientation of equipment.

Missiles are postulated, their trajectories are plotted, and their effects on safety systems evaluated. Where separation is not sufficient and where redundancy is not maintained, the missile is further investigated to determine whether it truly is a credible missile, and/or whether the impacted target can adequately withstand the impact. If the preceeding measures do not provide sufficient protection, then structural barriers are utilized to satisfy the safety requirements.

In addressing the credibility of valve bonnet missiles, a determination of the extent of reserve is made of the bonnet to valve connection. This includes:

1) The factor of safety of the valve test pressure over the operating design pressure.

2) The actual factor of safety of typical connections for the bonnet screwed on and retaining ring type bonnets.

3) The typical connection capabilities for the valve to bonnet connections assuming that a portion of the connection is not available to resist the design load.

Where this determination provides evidence of substantial reserve strength the credibility of the missile is discounted.

Where feasible, discreet orientation is prescribed for valve bonnets into a direction which precludes unacceptable damage from postulated missile action.

With regard to penetration of the containment, large valve components are addressed adhering to the general provisions for protection against missiles described above. In evaluating the credibility of large valve components, a determination is made of the extent of reserve strength that the valve component connection possesses in order to ascertain whether there is sufficient factor of safety to assure it will not become a missile. Again, where feasible, discreet orientation is prescribed for the valve so as to preclude its components from causing unacceptable damage to the containment. Where these procedures do not provide assurance that containment is not penetrated, protective barriers are provided.

The main steam safety relief valves and headers are postulated as possible missiles. Although postulated as credible missiles, based on the implementation of the criteria described above, they are found to have no effect on the shutdown capabilities of the plant, and no offsite radiological consequences.

Q. 211.017
(4.6.1)

Provide information demonstrating that the loss of the operating CRD pump at low reactor pressure (i.e., less than 500 psig) will not result in depressurization of the accumulator, thereby causing a loss of the reactor's scram capability. If the accumulator check valves were to leak following loss of the operating CRD pump, provide an estimate of the time interval before the reactor scram capability becomes marginal. In responding to this question, provide your basis for this estimate. Describe a test program or procedure which would provide assurance that operation of these check valves is acceptable over the lifetime of the WNP-2 facility.

Response:

The failure of a CRD pump will not affect the capability to scram all control rods if required. Scram is achieved on either HCU accumulator pressure or a combination of accumulator pressure and reactor pressure. Flow from the CRD pump is not required to successfully scram the plant. Each of the 185 control rod drives has its own HCU which operates independently of any others. 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 185 accumulators is detected by PSL-130 and LDS-129 respectively for each accumulator, as shown in Figure 4.6-5. Both occurrences are annunciated and a light signal identifies the particular control rod drive.

If a CRD pump fails the operator will bring the second pump on-line. If that pump is unavailable the operator can initiate a manual scram. If the pressure in a scram accumulator drops and approaches a pressure level below which control rod scram capability is impaired, an alarm is triggered and a light signal will identify the particular control rod drive. The operator will initiate a manual scram depending on the number of drives in this state.

If an accumulator check valve were to leak at the maximum allowable rate aginast which it has been designed, the minimum time available before scram capacity of an indi-

vidual drive becomes marginal is at least 20 minutes. This, however, does not mean that the total core scram capability becomes impaired due to the leakage from one check valve.

The core is designed to be shutdown from all operating conditions with the most reactive control rod fully withdrawn.

BWR reactor experience indicates there has been no failure to scram in over 200 reactor years that can be attributed to the reactor scram mechanical system of which the HCUs are a part.

No more than three failures of individual drives to scram have occurred in over 270,000 individual drive scrams.

Several failures to scram of individual drives would have to occur simultaneously to prevent reactor shutdown.

In summary, as previously mentioned, accumulator pressure is continuously monitored and a pressure decrease is alarmed to the operator, therefore, further analysis of the reliability and duration of the check valves to hold scram accumulator pressure is not needed.

Operational experience has shown that a testing program or procedure that would assure acceptable check valve operation is unnecessary.

The applicant's position is that it is unreasonable and unjustified to postulate simultaneously the loss of the CRD pump and, in addition, the standby CRD pump; the common mode failure of the accumulator check valves; and reactor pressure too low to drive the control rods into the reactor.

The events postulated utilize accident assumptions applied to normal operational events and assumes failure of non-safety grade equipment (CRD pump and CRD standby pump).

Q. 211.018
(4.6.1)

Experience at some operating BWRs indicates that failures can occur in the collet fingers of the CRD mechanism. In order to resolve this problem, some BWR facilities under construction have installed a revised collet retainer design. However, you do not address this particular problem in your FSAR nor do you discuss its resolution. Accordingly, confirm whether the revised collet retainer design will be incorporated into the CRD mechanisms of the WNP-2 facility. Revise Table 1.3-8 of the FSAR as required.

Response:

There have been no failures of collet fingers reported from the field, and no design change is contemplated. General Electric has demonstrated by testing and operating experience that the existing CRDs meet all safety and licensing requirements and are expected to give full life performances. However, as a result of examining operating drives, General Electric has discovered evidence of Intergranular Stress Corrosion Cracking (IGSCC) in some CRD drive components.

To preclude the potential for increased maintenance, we plan to incorporate the latest product improvements as provided by GE during our routine maintenance of the CRDs. Along with the other parts of the drive, the collet retainer tube, piston tube, and index tube will be routinely checked and changed out, if necessary, considering the latest design improvement available at the time.

Q. 211.019
(4.6.1)

We note in the third item of Table 1.3-8 of the FSAR that you intend to cut and cap the CRD return line as a resolution of the stress corrosion problem in this line. Discuss the impact of this modification on the plant. In particular, provide additional information addressing, but not limited to, the following items:

Part (a)

Compare the reactor vessel makeup capability when either one or two CRD pumps are operating, before and after the proposed modification. Additionally, provide a commitment to perform preoperational testing to verify the modified flow capability.

Part (b)

Provide a commitment to perform preoperational testing to verify the individual performance to the modified CRD components and those portions of the CRD system that are potentially affected by the cut and capped CRD return line (e.g., the equalizing valves, filters, scram times and the settling function).

Part (c)

If you choose to add new equalizing valves, discuss the potential effect on drive speeds throughout the lifetime of the WNP-2 facility. In particular, evaluate whether the CRD system can be adversely affected by a voiding of the drive exhaust header after a postulated single failure.

Part (d)

Evaluate the effect, throughout the lifetime of the WNP-2 facility, of the added flow through such components as the drive exhaust header and the stabilizing lines. In particular, discuss the increased potential for corrosion products from the carbon steel piping to be deposited as additional foreign matter in the drives.

Part (e)

Discuss the potential for, and effect on, flow reversal through the directional control solenoid valve over the plant lifetime.

Part (f)

Discuss the anticipated effect of your modifications to the CRD system on the P settling function across the control rod drives to ensure latching of the rods after withdrawal.

Response (a)

The calculated maximum impact on CRD flow injection capability resulting from the return line deletion is a reduction from approximately 165 gpm to approximately 135 gpm for single pump operation. The CRD system is not designed for two pump operation; no such design basis exists. The flow values presented were calculated assuming that all manual valves outside the primary containment, as well as all remote operated valves, have been adjusted to maximize the flow. Furthermore, in evaluating these flows, the RPV pressure was assumed to be up at the lowest set point of the safety/relief valves. Verification of CRD flow capability is part of the CRD preoperational test.

Response (b)

The control rod drive preoperational test will demonstrate that the system is fully operational and that all components including the hydraulic drive mechanisms, pumps, and flow control valves function properly. The CRD system will be configured with the modifications noted in the NRC concern.

Response (c)

In order to assure satisfactory system operation with the single failure of an equalizing valve, the proposed design modification will include the addition of two equalizing valves installed in a parallel configuration. The failure of either valve will not impair CRD operation for any foreseen operating or accident condition.

Response (d)

There will be no increased potential for carbon steel corrosion products to be deposited in the drives. All lines in the WNP-2 CRD Hydraulic System after the drive water filters, are made of stainless steel.

Response (e)

General Electric has completed lifetime testing of the subject directional control valves in response to the concern of pressurization and flow in the reverse direction. It is concluded from these tests that no adverse effects on the test valves resulted from the reverse flow mode of operation. (A copy of the report on these valve tests has been sent to Messrs. V. Stello and R. J. Mattson of the NRC by G. G. Sherwood of GE Licensing on April 9, 1979.)

Response (f)

In the new system configuration, the exhaust water header is essentially isolated from the rest of the CRD hydraulic system and maintained at nearly reactor pressure. During periods of rod motion and subsequent rod settling, the flow discharged from the drive to the exhaust water header is readily dissipated to adjacent drives (i.e., via reverse flow through the -121 directional control valves of adjacent HCUs) and causes the pressure in the exhaust water header to increase only a few psi. Thus no detrimental effects on rod settling performance is expected to result from this CRD system modification. Furthermore, evidence of satisfactory drive settling will be established during preoperational testing with the return line eliminated. CRD drive operation within acceptable defined margins will be demonstrated by this testing prior to plant operation.

Q. 211.020
(5.4.1)
(15.0)

Appendices G and H of the LaSalle and Zimmer FSARs, Respectively, provide information on the recirculation flow control system. State whether this information is applicable to the WNP-2 facility. If applicable, it should be referenced. Otherwise, provide comparable information regarding this system in your application.

Response:

Appendix H has been included in the WNP-2 FSAR as part of Amendment 4.

Q. 211.021
(5.4.1)

With respect to the recirculation flow control system:

a. Provide justification for the 8°F subcooling limitation in operating the recirculation pump.

b. In Section 5.4.1.3 of the FSAR, you state that if the subcooling falls below 8°F, the 60 Hz power supply is tripped to the 15 Hz power source to prevent cavitation of the recirculation pump, the jet pumps, and/or the flow control valve. This temperature limitation on subcooling appears to initiate a two-pump trip transient. Indicate whether the pump coastdown rate resulting from the above condition is more severe than the one assumed in the transient analysis in Chapter 15 of the FSAR. If so, reanalyze the pump trip transient using the more severe pump coastdown rate. Describe the consequences of a sudden increase in the recirculation pump speed which might occur, for example, due to an increase in the frequency of the power supply.

Response:

a. Recirculation system computer analysis shows that for core flows beyond the range of applicability of the flow control valve (FCV) cavitation interlock, jet pump nozzle cavitation is the most limiting of the recirculation system component cavitation concerns. The jet pump nozzle is calculated to enter incipient cavitation at subcoolings below 10°F (8°F is not correct for WNP-2 with the FCV wide open, which is the most limiting location on the power/flow map). The 10°F setpoint, therefore, protects against recirculation system cavitation for all core flows beyond the range that the FCV cavitation interlock operates. Page 5.4-3 is being changed to show 10°F vice 8°F.*

*Draft page changes attached.

b. The condition considered in the transient analysis in Chapter 15 is the most severe as regards pump coastdown rate; no reanalysis is necessary. The final sentence in Question 211.021b is concerned with the potential for recirculation system cavitation following a two-pump trip when speed is increased slightly relative to the LFMG set speed. The margin to cavitation would be a function of the amount of speed increase, but in no case would the speed induce cavitation because of the relatively large degree of subcooling available when operating on the LFMG set.

When the pump is operating at 100% speed most of the NPSH is supplied by the subcooling provided by the feedwater flow. Accurate temperature detectors* are provided in the recirculation lines and the steam lines beyond the second isolation valves. The difference between these two readings is a direct measurement of the subcooling. If the subcooling falls below 8°F the 60 Hz power supply is tripped to the 15 Hz power source to prevent cavitation of recirculation pump, jet pumps, and/or the flow control valve.

10°F

When preparing for hydrostatic tests, the nuclear system temperature must be raised above the vessel nil ductility transition temperature limit. The vessel is heated by core decay heat and/or by operating the recirculation pumps at 100% speed.

Connections to the piping on the suction and discharge sides of the pumps, as shown on Figure 5.4-2 provide a means to flush and decontaminate the pump and adjacent piping. The piping low point drain, designed for the connection of temporary piping, is used during flushing or decontamination.

Each recirculation pump is driven by a constant speed motor and is equiped with mechanical shaft seal assemblies. The two seals built into a cartridge can be readily replaced without removing the motor from the pump. Each individual seal in the cartridge is designed for pump design pressure so that any one seal can adequately limit leakage in the event that the other seal should fail. The pump shaft passes through a breakdown bushing in the pump casing to reduce leakage in the event of a gross failure of both shaft seals. The cavity temperature and pressure drop across each individual seal can be monitored.

*The system accuracy of the temperature detectors, signal processing equipment and bistable devices is ±0.2°F at the 95% (two standard deviation) confidence level.

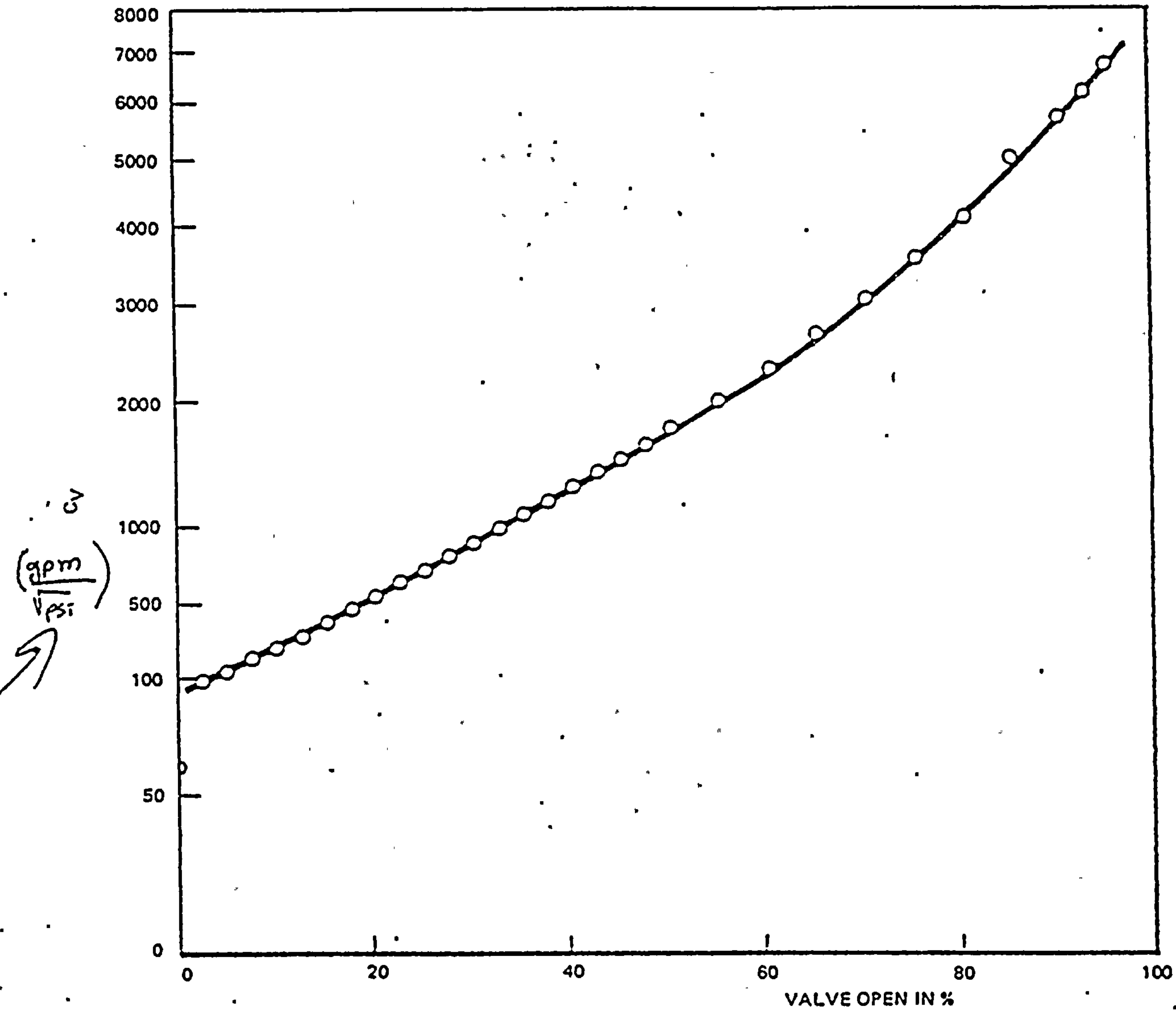
Q. 211.022
(5.4.1)

Indicate the units associated with the value of C_V shown in Figures 5.4-4a and 5.4-4b of the FSAR.

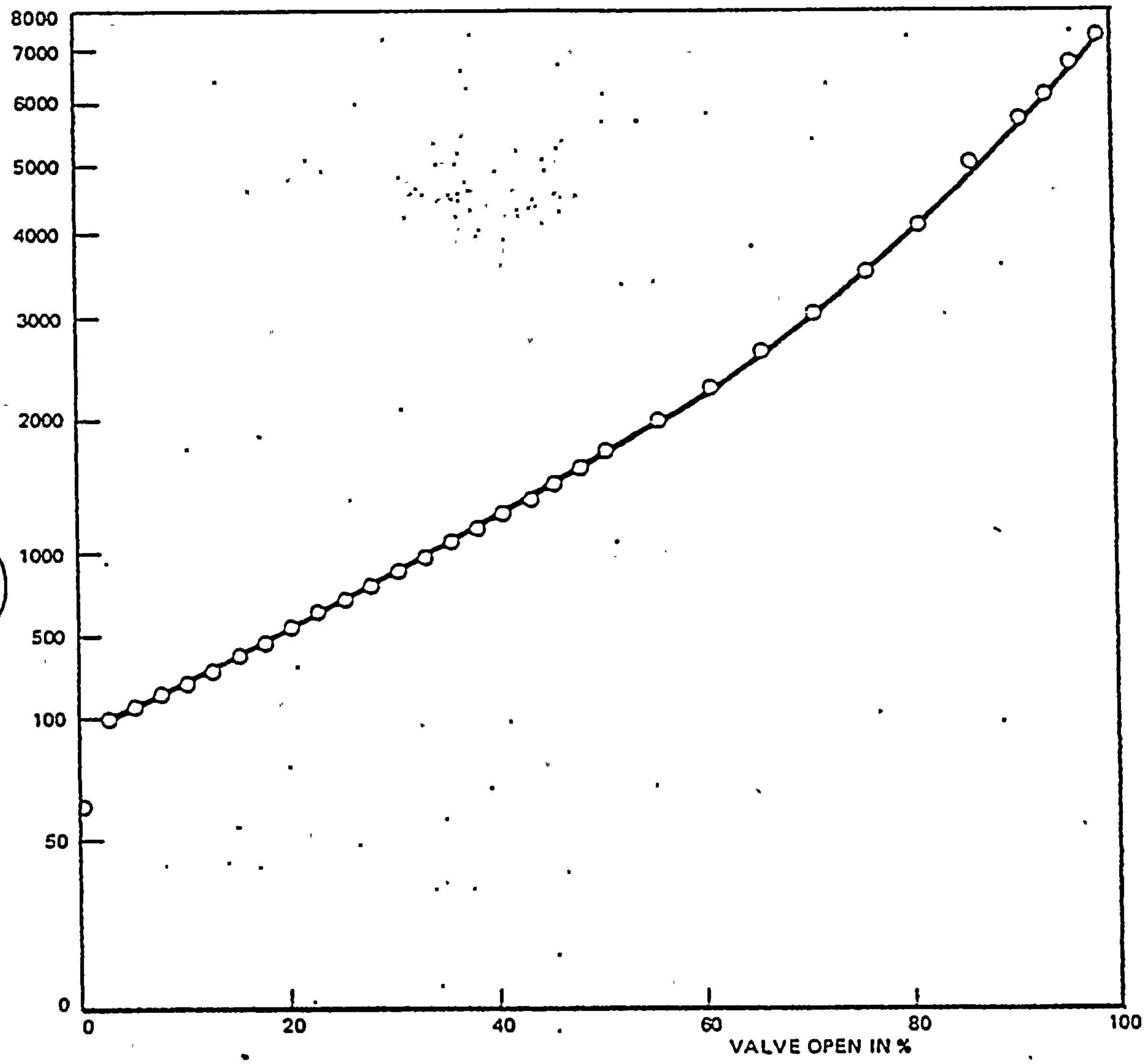
Response:

The units of FCV C_V are gpm/$\sqrt{psi}$.*

*Draft figure changes attached.



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	FLOW CONTROL VALVE CHARACTERISTIC CLOSING FROM (100% FLOW - 0% FLOW)	FIGURE 5.4-4a



WASHINGTON PUBLIC POWER SUPPLY SYSTEM NUCLEAR PROJECT NO. 2	FLOW CONTROL VALVE CHARACTERISTIC OPENING FROM (0% FLOW - 100% FLOW)	FIGURE 5.4-4b

Q. 211.023
(3.5)
(4.6.1)

Provide assurance that the essential portions of the CRD system (i.e., the 1-inch supply and return piping located inside the containment) shown in Figures 3.5-20, 3.5-22, and 3.5-27 of the FSAR, are protected from the effects of high or moderate energy line breaks. In responding to this item, consider postulated breaks such as ones in the high pressure core spray (HPCS) system, the feedwater injection system and the reactor coolant pressure boundary. Our concern is whether pipe whip and/or jet impingement forces resulting from these postulated breaks can impair the capability of the CRD system to scram. Additionally, provide an assessment of the damage which could occur to the cluster of CRD return and supply lines, including the effect on the scram capability, due to a postulated rupture of a single CRD supply or return line.

Response:

Rupture of CRD supply and return piping inside containment would not prevent safe shutdown since the control rods would then be inserted by reactor pressure. The only accident affecting CRD piping inside containment which could prevent safe shutdown would involve crimping of more than one 3/4 inch return line in any 3 x 3 array of control rods such that reactor pressure could not insert the control rods. A testing program has been initiated to evaluate the susceptibility of the 3/4 inch CRD piping to crimping caused by pipe break accidents inside containment. The complete response to this question will be provided later after the testing program has been completed.

Q. 211.024
(5.4.7)

It is our position for all light-water-reactors that the RHR system shall be capable of bringing the reactor to a cold shutdown condition using only safety-grade systems. Confirm that this requirement is satisfied for the WNP-2 facility. In responding to this request, include a consideration of the capability of the air supply system which is used to operate the RCIC steam and condensate control valves located at the RHR heat exchanger, when the RHR system is in the steam condensing mode.

Response:

All portions of the RHR system required to function in bringing the reactor to a cold shutdown condition are safety grade and redundant except for the shutdown cooling suction line. If this line were unavailable due to a single failure of a suction valve, a safety grade alternate shutdown cooling path can be established through the ADS valves as described in the notes to Figure 15.2-11, Activity C1 or C2.

The steam condensing mode is used only to maintain hot standby condition should the vessel be isolated from the main condenser. Specifically, it allows for maintenance on the turbine generator set without first requiring a cold shutdown of the RPV or continued opening of the main steam relief valves to the suppression pool.

No analysis has been performed which demonstrates that the steam condensing can be used to bring the reactor to a safe, cold shutdown. No credit has been taken for the steam condensing mode in any safety analysis, accordingly, it is permissible to use non-safety air for E12-F051 (RHR-PCV-51) and E12-F065 (RHR-LCV-65). On a loss of air these valves fail-shut, the desired position during accident conditions.

Q. 211.025

It is also our position for all light-water reactors that the RHR system shall be capable of bringing the reactor to a cold shutdown condition with only on-site or off-site power available, assuming the most limiting single failure. In this regard, while we note that Figure 15.2-10 of the FSAR shows a number of available success paths to achieve a cold shutdown condition, vessel depressurization using the RHR system in the steam condensing mode is not shown. (This latter mode is one of the success paths when off-site power is not available.) Either correct this figure or justify this omission. If vessel depressurization were to be achieved by manual actuation of the relief valve, indicate how many valves would have to be actuated. Describe your plans for testing the alternate modes to achieve shutdown cooling. Demonstrate that adequate passage of water through the safety/relief valves can be achieved and maintained when the alternate method is in use. Indicate the quantity of air supplied, its source, and the time interval before the air is exhausted.

Response:

The omission of the steam condensing mode is justified because there is no requirement for the steam condensing mode to be used to bring the reactor to a cold shutdown. Steam condensing is not a safety grade means to depressurize the reactor.

If vessel depressurization were to be achieved by manual actuation of relief valves, three valves would need to be actuated to pass sufficient steam flow to depressurize the vessel. Three to five valves would be necessary to pass sufficient water to keep the vessel depressurized as necessary. Testing of single phase liquid or two phase flow through the relief valves at low pressure conditions will be performed under the auspices of the BWR Owners Group in response to NUREG-0578 item 2.1.2. The details of this test program will be presented to the NRC by the Owners Group in early 1980.

The air supply for the ADS valves is discussed in the response to Question 211.048.

Q. 211.026
(5.4.7)

In the shutdown cooling mode, the flush water valves are opened and closed outside the control room. Identify in Section 5.4.7.2 of the FSAR, the local flush water valves which are operated and the source of this flush water. Discuss the consequences if the operator were to omit this procedure and/or forgot to close a local flush water valve and continue shutdown operations. Include a discussion of the available interlocks in your response.

Response:

The only valve which will be opened for flushing or filling and then closed is F007 (Reference Figure 5.4-13a). F007 is opened to assure that the suction line is filled. F040 and F049 are opened to direct the prewarming/flushing water to radwaste (Reference revised Section 5.4.7.2.6)*.

If the operator omitted the shutdown cooling mode flush, the lower quality water which has been laid up in the RHR lines would be pumped into the reactor. Additional reactor water cleanup would be required to attain acceptable reactor water quality.

If the operator forgot to close F007 and continued shutdown procedures, reactor coolant could possibly pressurize the condensate supply system up to 213 psig, which is above the 150 psig design pressure. Therefore, a check valve will be added to the condensate supply line to prevent this. The maximum pressure is limited by the maximum static head between the reactor water level and the lowest portion of the condensate supply system and the 135 psig interlock which will close the shutdown cooling suction valves F008 and F009.

F040 and F049, which direct the prewarm/flushing water to radwaste, are operated from the control room and are provided with position indication so it is highly unlikely they would be left open after prewarming. If they were accidently left open after prewarming, the loss of water from the RHR system would be detected by a high tank level alarm in the radwaste system or a low pressure alarm in the RHR discharge pipe from PIS-N022.

*Revised draft page changes attached.

Transients are treated by items a and c; item b above results from an excessive leak past isolation valves. E12-F055 and RHR-RV-95 are sized to maintain upstream piping at 500 psig and 10 percent accumulation with E12-F051 (steam inlet) or E12-F087 fully open and a reactor pressure equal to the lowest Nuclear Boiler safety/relief valve spring set point. E12-F036 are sized to maintain upstream pressure at 75 psig and 10 percent accumulation with both PCV E12-F065 A&B RCIC suction failed open. E12-F005, F025, F088, and F030 are set at the design pressure specified in the process data drawing plus 10 percent accumulation.

Redundant interlocks prevent opening valves to the low pressure suction piping when the reactor pressure is above the shutdown range. These same interlocks initiate valve closure on increasing reactor pressure.

A pressure interlock prevents connecting the discharge piping to the primary system whenever the pressure difference across the discharge valve is greater than the design differential. In addition a high pressure check valve will close to prevent reverse flow if the pressure should increase. Relief valves in the discharge piping shall be sized to account for leakage past the check valve.

5.4.7.1.4 Design Basis With Respect to General Design Criteria 5

The RHR system for this unit does not share equipment or structures with any other nuclear unit.

5.4.7.1.5 Design Basis for Reliability and Operability

The design basis for the Shutdown Cooling mode of the RHR system is that this mode is controlled by the operator from the control room. The only operation~~s~~ performed outside of the control room for a normal shutdown ~~are~~ is manual operation of a local flushing water admission valve~~s~~, which ~~are~~ is the means of ~~providing clean water to~~ ASSURING THAT THE SUCTION LINE OF the shutdown portions of the RHR system IS FILLED AND VENTED.

Two separate shutdown cooling loops are provided; and although both loops are required for shutdown under normal circumstances, the reactor coolant can be brought to 212°F in less than 20 hours with only one loop in operation. With the exception of the shutdown suction, shutdown return, and steam supply and condensate discharge lines, the entire RHR system is part of the ECCS and containment cooling systems, and is therefore designed with redundancy, flooding protection, piping protection, power separation, etc. required of such

JEL 10/4/80

5.4.7.2.4 Applicable Codes and Classifications

Refer to 3.2.

5.4.7.2.5 Reliability Considerations

The Residual Heat Removal System has included the redundancy requirements of 5.4.7.1.5. Two completely redundant loops have been provided to remove residual heat, each powered from a separate emergency bus. With the exception of the common shutdown line, all mechanical and electrical components are separate. Either loop is capable of shutting down the reactor within a reasonable length of time.

5.4.7.2.6 Manual Action

a. Residual Heat Removal (Shutdown Cooling Mode)

VALVE E12 F007 IS OPENED TO FILL THE SUCTION LINE, AFTER WHICH IT IS CLOSED

VIA VALVES E12-F040 AND F049 WHICH ARE OPERATED FROM THE CONTROL ROOM

In shutdown operation, when vessel pressure is 135 psig or less, the pool suction valve is closed for the ~~initial~~ S shutdown loop. ~~Flushing valves are opened, and the stagnant water flushed to radwaste via valves E12-F040 and F049 which are operated from the control room. At the end of this nominal flush,~~ the testable check bypass valve may be opened in the shutdown return line and vessel water is permitted to enter the upper portion of the ~~chosen~~ S loop to prewarm it AND PROVIDE A NOMINAL FLUSH. Effluent is directed to radwaste and a temperature element is used to control effluent temperature. The testable check bypass valve is closed and vessel suction valves are opened to allow prewarming of the lower half of the shutdown loop with effluent directed to radwaste as before. The radwaste effluent valves are closed, ~~the heat exchanger bypass valves opened (the exchanger valves were closed after the initial cold water flush),~~ then the RHR pump IS start~~s~~ED at a regulated flow through return valve E12-F05~~2~~3. ~~After waiting several minutes to permit loop internal stability to be established,~~ the service water pump is started AND the service water valves are opened. ~~the heat exchanger inlet and outlet valves are opened~~ and cooldown of the vessel is in progress. Cooldown rate is subsequently controlled via valve E12-F053 (total flow) and E12-F048 (heat exchanger bypass flow). All operations are performed from the control room except for opening and closing of ~~local flush water valves~~ F007.

GAD 10/6/79

Q. 211.027
(5.4.7)

In Section 5.4.7.1.3 of the FSAR, you indicate the specific RHR relief valves and the RHR design pressures used as the basis for providing relief capacity. Expand your discussion by indicating the relief valve capacity, the nominal setpoints, the setpoint tolerance, and the ASME class designation of these valves and lines. In addition, discuss the vulnerability of the RHR system to malfunctions which could result in overpressurization of low pressure piping. Support your evaluation by providing an outline of all operating procedures required to bring the plant to a cold shutdown condition from hot standby and the procedures for plant startup from cold shutdown.

Response:

The safety relief valves protecting the RHR system are listed below (Reference Figures 5.4-13a and 5.4-13b):

Relief Valve	Nominal Setpoint/Capacity	Location	Piping Design Pressure
F088	125 psig/10 gpm	RHR pump suction from suppression pool	125 psig (Loop C) 220 psig (Loops A and B)
F005	220 psig/25 gpm	RHR pump suction from recirc pipe	220 psig
F025	500 psig/25 gpm	RHR discharge	500 psig
F030	125 psig/10 gpm	RHR flush line to radwaste	125 psig
F036	75 psig/1750 gpm	RHR HX condensate to suppression pool or RCIC pump suction	125 psig
F055	500 psig/330,000 lbs/hr	Steam supply to RHR heat exchanger	500 psig

Relief Valve	Nominal Stepoint/Capacity	Location	Piping Design Pressure
RHR-RV-95*	500 psig/330,000 lbs/hr	Steam supply to RHR heat exchanger	500 psig

*RHR-RV-95 is currently not shown in Figures 5.4-13a and 5.4-13b.but is shown on Figure 3.2-6, Zones E, 13 and E, 4.

All safety relief valves are purchased to ASME III, Class 2 requirements to match the requirements of the piping they are protecting. As such, the setpoint tolerance is + 3%, per ASME III, Section NC-7614.2.

The RHR system is connected to higher pressure piping at: (1) shutdown suction; (2) shutdown return; (3) LPCI injection; (4) head spray; and (5) heat exchanger steam supply. The vulnerability to overpressurization of each location is discussed in the following paragraphs.

Shutdown suction has two gate valves (F008 and F009) in series which have independent pressure interlocks to prevent opening at high inboard pressure (135 psig reactor pressure). No single active failure or operator error will result in overpressurization of the lower pressure piping. With the RHR pumps normally lined up to the suppression pool (F006 closed), the shutdown cooling suction line is protected for thermal expansion or from leakage past F008 and F009 by F005. With all the RHR suction valves closed, the suction piping is protected for thermal expansion or leakage past the discharge check valves by F088.

The shutdown return line has a swing check valve (F050) to protect it from higher vessel pressures. Additionally, a globe valve (F053) is located in series and has a pressure interlock to prevent opening at high inboard pressures (135 psig reactor pressure). No single active failure or operator error will result in overpressurization of the lower pressure piping.

The LPCI injection line has an air testable swing check valve (F041) to protect it from higher vessel pressures. The air operator on the testable check valve is only capable of opening the testable check valve if the differential pressure is less than 2.0 psid. Additionally,

a gate valve (F042) is located in series and has pressure interlocks to prevent opening at high differential pressure (nominally 750 psid). No single active failure or operator error will result in overpressurization of the lower pressure piping.

The head spray piping has three swing check valves in series (two belonging to the RCIC system and one (F019) belonging to the RHR system), to protect it from higher vessel pressures. Two of the swing check valves have air operators but they are only capable of opening the testable check valve if the differential pressure is less than 2.0 psid. Additionally, a globe valve (F023) is located in series and has a pressure interlock to prevent opening at high inboard pressures (135 psig reactor pressure). No single active failure or operator error will result in the overpressurization of the lower pressure piping.

Overpressurization protection of the RHR discharge piping for thermal expansion or from leakage past the head spray, shutdown injection, and LPCI isolation valves is provided by F025.

The heat exchanger steam supply line has a globe valve (F052) for shutoff. The operator admits steam through F052 and sets the pressure regulating valve (F051) to limit heat exchanger pressure to about 200 psig. Also, F087 can be opened when the steam supply pressure is below the pressure interlock (500 psig) to provide additional steam flow rate to the heat exchangers. Two relief valves (F055) and RHR-RV-95 with a combined capacity of 660,000 lbs/hr are provided downstream of F051 and F087 to protect the low pressure piping should F051 fail open. The maximum calculated steam flow rate (sonic flow) with F051 and F052 failed open is 600,000 lbs/hr, so there is adequate relief valve capacity to handle this failure. The Class 1E lear detection system, which monitors steam flow rate to the RHR heat exchangers, will isolate the steam supply (close F076, F063 and F064 per Figure 5.4-9a) when the steam flow reaches approximately 360,000 lbs/hr (175% decay heat steam generation rate 1/2 hour after scram). No single active failure nor operator error will cause overpressurization of the lower pressure piping.

During steam condensing mode, with the RHR heat exchanger at 200 psig, the condensate is dumped to either the suppression pool or the RCIC pump suction. F036 provides protection to this low pressure piping should both level control valves F065A and F065B fail open.

F030 protects the drain piping from the RHR system to radwaste from thermal expansion or from leakage past the isolation valves F071 and F072.

OUTLINE OF OPERATING PROCEDURE AND RHR OVERPRESSURIZATION SAFEGUARDS

1. Plant Shutdown to Cold Shutdown from Hot Standby* With Safety Grade Systems

Reactor Condition	Operating Mode Used	RHR Overpressurization Safeguard
Depressurization from hot standby to 135 psig	o Main steam relief valve discharge to the suppression pool depressurizes vessel	RHR isolated.
	o Initiate and operate pool cooling mode of RHR system	Low pressure mode, no safeguard required.
Cooldown from 135 psig to cold shutdown	o Initiate and operate shutdown cooling mode of RHR	Redundant pressure interlocks on F008 and F009 close valve above pressure interlock setpoint.

2. Plant Startup from Cold Shutdown

Reactor Condition	Operating Mode Used	RHR Overpressurization Safeguard
Reactor coolant below 125°F and RPV head replaced	o Terminate shutdown cooling and isolate RHR	Redundant pressure interlocks on F008 and F009 close valves above pressure interlock setpoint.
Remainder of startup	o Standard	RHR isolated.

*Normally, the main condenser is the heat sink during hot standby, but, because of larger RHR interface, it is assumed that the main condenser is unavailable.

Q. 211.028
(5.4.7)

Discuss the need and provide the design basis for incorporating a pressure interlock to prevent the connection of the RHR discharge piping to the primary system whenever the actual pressure difference across the discharge valve is greater than the design value for this pressure differential. Identify the affected valves.

Response:

Refer to Figure 5.4-13a for valve numbers. The RHR discharge piping is connected to higher pressure piping at shutdown return, LPCI injection and head spray. Only the LPCI injection valve (F042) has a pressure differential interlock. The other two injection points have reactor pressure interlocks. The interlocks are described below.

The shutdown return line has a swing check valve (F050) to protect it from higher vessel pressures. Additionally, a globe valve (F053) is located in series and has a pressure interlock to prevent opening at high inboard pressures (approximately 135 psig reactor pressure), which is well below the design pressure of the RHR discharge pipe (500 psig). No single active failure or operator error will result in overpressurization of the lower pressure piping.

The LPCI injection line has an air-testable swing check valve (F041) to protect it from higher vessel pressures. Additionally, a gate valve (F042) is located in series and has a pressure interlock to prevent opening at high differential pressure (approximately 700 psid). The pressure differential setting is based on the difference between full reactor pressure and the shutoff head of the RHR pump. This allows the injection valve to open at the earliest possible moment after a LOCA signal. Although the injection valve (F042) is open, the RHR discharge pipe is not subject to primary system pressure until the RHR pump discharge pressure exceeds reactor pressure and opens the check valve (F041). This will occur at about 300 psig, which is the approximate shutoff head of the RHR pump. No single active failure or operator error will result in overpressurization of the lower pressure piping.

The head spray piping has three swing check valves in series (two belonging to the RCIC system and one (F019) belonging to the RHR system), to protect it from higher vessel pressures. Additionally, a globe valve (F023) is located in series and has a pressure interlock to prevent opening at high inboard pressures (approximately 135 psig reactor pressure), which is well below the design pressure of the RHR discharge pipe (500 psig). No single active failure nor operator error will result in overpressurization of the lower pressure piping.

Q. 211.029
(5.4.7)

Provide more detailed information in Section 5.4.7.1 of the FSAR regarding the actuation of the automatic minimum flow valves which are used to protect the RHR pumps from damage if these pumps were to be operated when the discharge valve is closed. For example, state the flow rates which would initiate a signal to open and close the minimum flow valves. Indicate whether the control system satisfies the requirements of IEEE Standard 279-1971.

Response:

The minimum flow valve opens at main line flows less than 550 gpm. This allows flow to return to the suppression pool through the low resistance low flow bypass line, which branches off the main line upstream of the flow element. The minimum flow valve closes at main line flows greater than 550 gpm and forces the entire pump discharge flow through the main line. The text of 5.4.7.1.2 is revised to reflect this.*

The minimum flow valve controls meet IEEE-279 requirements.

*See attached draft page changes.

5.4.7.1.1.3 Suppression Pool Cooling Mode

The functional design basis for the suppression pool cooling mode is that it shall have the capacity to ensure that the suppression pool temperature immediately after a blowdown shall not exceed 170°F.

5.4.7.1.1.4 Containment Spray Cooling Mode

The functional design basis for the containment spray cooling mode is that there should be two redundant means to spray into the drywell and suppression pool vapor space to reduce internal pressure to below design limits.

5.4.7.1.1.5 Reactor Steam Condensing Mode

The functional design basis for the reactor steam condensing mode is that the heat exchanger in one loop of the RHR system, in conjunction with the RCIC turbine, shall be able to condense all of the steam generated after a reactor scram 1-1/2 hours after scram.

5.4.7.1.2 Design Basis for Isolation of RHR System from Reactor Coolant System

The low pressure portions of the RHR system are isolated from full reactor pressure whenever the primary system pressure is above the RHR system design pressure. See 5.4.7.1.3 for further details. In addition, automatic isolation may occur for reasons of vessel water inventory retention which is unrelated to line pressure rates. (See 5.2.5 for an explanation of the Leak Detection System and the isolation signals.)

The RHR pumps are protected against damage from a closed discharge valve by means of automatic minimum flow valves, WHEN THE which open ~~on low~~ main line flow and close ~~on high~~ main line flow
WHEN THE IS LESS THAN 550gpm
IS GREATER THAN 550 gpm.

5.4.7.1.3 Design Basis for Pressure Relief Capacity

The relief valves in the RHR system are sized on one of three bases:

a. Thermal relief only

b. Valve bypass leakage only

c. Control valve failure and the subsequent uncontrolled flow which results.

QSI 7/30/79

Q. 211.030
(5.4.7)

In Figure 5.4-15 of the FSAR you present the RHR pump characteristic curves. However, two sets of curves are shown, one for the maximum diameter piping and the other for the minimum diameter piping. Indicate which of the head versus flow rate characteristics was used in the performance evaluation of the ECCS and the RHR system.

Response:

Figure 5.4-15 is being replaced with Figures 5.4-15 a, b, and c. These figures are the actual pump performance curves.

The LPCI flow assumed in the ECCS analysis is given in Table 6.3-2. Percent LPCI flow versus reactor pressure vessel (RPV) pressure is given in Figure 6.3-9. RHR flows assumed in the containment analyses for other modes of RHR operation are given in Table 6.2-2. Calculations performed by the AE using the pump performance curves ensure the flow values in Figure 5.4-14b and in the "full capacity" column of Table 6.2-2 are met.*

*Draft FSAR page changes attached.

5.4.7.2.2 Equipment and Component Description

a. System Main Pumps

The RHR main system pumps are motor-driven deep-well pumps with mechanical seals and cyclone separators. The motors are water cooled. The pumps are sized on the basis of the LPCI mode (Mode A) and the minimum flow mode (Mode G) of the Process Data Figure 5.4-14b. Design pressure for the pump suction structure is 220 psig with a temperature range from 40 to 360°F. Design pressure for the pump discharge structure is 500 psig. The bases for the design temperature and pressure are maximum shutdown cut-in pressures and temperature, minimum ambient temperature, and maximum shutoff head. The pump pressure vessel is carbon steel, the shaft is stainless steel. A comparison between the required NPSH (obtained from the pump characteristic curves provided in Figures ~~5.4-15)~~ and the NPSH needed in the Process Diagram Figure 5.4-14b (Note 8) demonstrates the required NPSH is adequate. Available NPSH is calculated per Regulatory Guide 1.1.

stet

6.3-10a,b

b. Heat Exchangers

The RHR system heat exchangers are sized on the basis of the duty for the shutdown cooling mode (Mode E of the Process Data). All other uses of these exchangers, including steam condensing, require less cooling surface.

Flow rates are 7450 gpm (rated) on the shell side and 7400 gpm (rated) on the tube side (service water side). Rated inlet temperature is 125°F shell side and 85°F tube side. The overall heat transfer coefficient is 195 BTU per hour square foot. The exchangers contain 7670 ft^2 of effective surface. Design temperature range of both shell and tube sides are 40°F to 480°F. Design pressure is 500 psig on both sides. Fouling factors are 0.0005 shell side and 0.0002 tube side. The construction materials are carbon steel for the pressure vessel with stainless steel tubes and stainless steel clad tube sheet.

TABLE 6.2-2 (Continued)

REVISED INPUT

	Full Capacity	Containment Analysis Value Case A	Case B	Case C
C. CONTAINMENT COOLING SYSTEM:				
1. Number of Pumps	2	2	1	1
2. Pump Capacity, gpm/pump	7450	7067*	7067*	7067*
3. Heat Exchangers RHR System-Inverted U-tube, single pass Shell, Multi-pass Tubes, Vertical Mounting				
a. Number	2	2	1	1
b. Heat Transfer Area, ft^2/Unit	7641	7641	7641	7641
c. Overall Heat Transfer Coefficient, Btu/hr-ft^2-^oF	195 (fouled) 400 (clean)	195	195	195
d. Standby Service Water Flowrate per Exchanger, gpm	7400	7400	7400	7400
e. Design Service Water Temperature Minimum, ^oF Maximum, ^oF	 32^oF 85^oF	 95*	 95*	 95*

TABLE 6.2-2 (Continued)

	Full Capacity	Containment Analysis Value Case A	Case B	Case C
D.3 (Cont'd)				
3. Low Pressure Coolant Injection (LPCI)				
a. No. of Pumps	3	3	1	1
b. No. of Lines	3	3	1	1
c. Flowrate, gpm/line	7450**	7067*	7067*	7067*
4. Residual Heat Removal (RHR)				
a. Pump Flowrate:				
Shell Side	7450			
Tube-Side	7400			
b. Source of Cooling Water		Standby Service Water (SSW)		
E. AUTOMATIC DEPRESSURIZATION SYSTEM				
1. Total Number of Safety/ Relief Valves	18			
2. No. actuated on ADS	7			

REVISED INPUT

* Represents conservative value used in analysis.
** Increases to 7900 gpm with zero differential pressure between RPV and wetwell.

Q. 211.031
(5.4.7)

In Table 5.4-3 of the FSAR, you indicate that the RHR isolation valves MOF008 and MOF009 are closed upon generation of a signal indicating reactor low water level. It appears that you have mislabelled these valves in this table as "recirculation line suction" rather than as "RHR isolation". Indicate whether this valve isolation signal is based on the same signal as the RHR pump actuation in the low pressure coolant injection system (LPCI) mode (i.e., a water level which is 1.0 foot above the active core). If not, indicate the water level in the reactor pressure vessel at which the isolation signal is generated, thereby isolating the RHR suction valves. Show that cooling of the reactor core can be maintained assuming a pipe break outside the containment. Assuming a pipe break outside containment in the RHR system when the plant is in a shutdown cooling mode, provide the following additional information:

a. Identify the systems available for maintaining core cooling.

b. Indicate the maximum discharge rate resulting from the postulated break and the time interval available for recovery based on the discharge rate and its effect on core cooling.

c. Identify the alarms available to alert the operator in the event of such a break and show that sufficient time is available for operator action to prevent damage to safety-related systems.

d. Indicate what recovery procedures are available.

e. Following a postulated break in a moderate energy line, the single failure criterion should be applied in the manner discussed in Section 3.6.1 of the Standard Review Plan (SRP) NUREG-75/087, and in Branch Technical Position APCSB 3-1, "Protection Against Postulated Piping Failures In Fluid Systems Outside Containment", November 24, 1975.

Response:

a) F008 and F009 isolate at reactor water level 3 which is 180 inches above the top of the active fuel. LPCI is initiated at reactor water level 1 (Reference FSAR Figure 5.2-6.) Should a pipe failure occur, outside the containment, in the RHR system when the plant is in shutdown cooling, acceptable core cooling would be achieved by the core cooling systems. The following core cooling systems would be available to maintain core cooling when applying SRP 3.6.1 and BTP APCSB 3-1:

- If the single active failure is HPCS the following are available: LPCS 2 LPCI.
- If the single active failure is LPCS the following are available: HPCS 2 LPCI.
- If the single active failure is LPCI (not shutdown cooling loop) the following are available: HPCS LPCS 1 LPCI.

b) The maximum discharge resulting from the largest crack in the RHR piping outside containment is determined using the guidelines in BTP MEB 3-1 for moderate energy piping. The maximum discharge rate is estimated to be 1000 gpm (to be confirmed later in the ongoing pipe break and missile study). This is based upon a pipe break in the pump discharge piping (20" schedule 30) at the pump discharge flange, normal water level in the reactor during shutdown cooling (approximately 50 inches below the steam line nozzles), reactor pressure of 135 psig, and the RHR pump running at 7450 gpm (normal shutdown flow). See "c" below for the time interval available for recovery.

c) The following alarms are available to the operator in the event of a pipe break in the shutdown cooling line outside containment.

1. Low reactor water level alarm (Level 4, 198.7 inches above active fuel).
2. Low reactor water level (Level 3) to scram and isolate MOF008 and MOF009.
3. Equipment area high temperature (Class 1E) to isolate MOF008 and MOF009.

4. High flow rate in the shutdown cooling suction line to isolate MOF008 and MOF009 (Class 1E).

5. Reactor building floor drain sump level for leakage rates greater than 50 gpm.

6. Reactor building floor drain leakage rate alarm (5 gpm).

7. ECCS pump room flood level instrumentation (Class 1E), installed to detect passive failures in the ECCS post-LOCA (Reference response to FSAR Question 212.003).

Notwithstanding these alarms, however, only about 13,000 gallons of water will spill out the break before the reactor water level drops from the normal shutdown cooling level to Level 3, automatically closing MOF008 and MOF009 and isolating the break. No single active failure can prevent isolation of the break. During the time of the pipe break, HPCS will not automatically initiate, because its initiation signal (Level 2) is about 50 inches below Level 3. RCIC will not automatically initiate because the reactor pressure (135 psig maximum during shutdown cooling) is too low to operate the RCIC turbine.

For the largest pipe break (1000 gpm), which can only occur in the RHR A or B pump rooms, the flood level resulting from 13,000 gallons will not affect operation of either RHR pump A or B. In addition, the flooding would only affect the room in which the pipe break occurred because of the water tight integrity of the RHR A and B pump rooms. Therefore, no operator action is required to protect these pumps.

Pipe breaks in the RHR shutdown cooling mode which can affect other ECCS systems (LPCS, HPCS or RHRC) via flooding through the floor drain piping in the upper portions of the reactor building will have flooding rates less than 1000 gpm because they will be higher in the building (less static head), have less driving head due to friction losses, and smaller crack sizes (smaller diameter pipe). These pipe breaks in the upper portion of the reactor building will be immediately detected by the high flow (5 gpm) alarms in the floor drain downcomer piping. Regardless of what the pipe

break discharge rate is, the flood level resulting from 13,000 gallons is not capable of affecting the operation of the LPCS, HPCS or RHRC pumps, assuming all of the water is spilled into each pump room. Again, no operator action is required to protect these pumps.

It should be noted that the environmental effects (pressure, temperature and humidity) of pipe breaks during shutdown cooling ~~are~~ being addressed by ongoing pipe break and missile study.

d) Core cooling is not a concern. Sufficient ECCS equipment remains functional to automatically keep the core covered at all times. The cold shutdown procedure, i.e., containment heat removal, needs to be resumed. If the pipe break disables the common shutdown cooldown suction line, cold shutdown can be assumed by the alternate shutdown cooldown path discussed in Section 15.2.9. If the pipe break in the shutdown cooldown line is downstream of the F006 valve, normal shutdown cooldown can be resumed using the redundant RHR shutdown loop.

e) For application of single failure criteria. See "a" above.

Q. 211.032
(5.4.7)

Discuss the system design provisions incorporated into the facility to prevent damage to the RHR pumps in the LPCI mode during pump runout conditions in the ECCS operating mode and in the test mode. We note that Figures 5.4-13a, 5.4-13b and 5.4-14a of the FSAR indicate that a metering orifice is installed in the discharge lines. Indicate whether this metering orifice can perform the same function as a restricting orifice. If not, it is our position that the discharge lines of the RHR pumps should incorporate a restricting orifice.

Response:

The metering orifice in the discharge line does not serve as a restricting orifice.

The piping for each mode of RHR operation is investigated to ensure that the resistance is low enough to allow the rated flows given in Figure 5.4-14b yet high enough to prevent pump runout. Restricting orifices are necessary in the system test lines (for suppression pool cooling and test modes). Engineering changes are currently being processed which will add these restricting orifices. Restricting orifices in the LPCI lines are currently being investigated. The balance of the RHR modes do not require restricting orifices in their lines.

Q. 211.033
(5.4.7)

Provide a more detailed description, including the location, of the RHR pump suction strainer which is inside the suppression pool. Indicate the pipe bends and the minimum height of the suppression pool water level above this strainer. Show that the required net positive suction head (NPSH) at the centerline of the RHR pump will be available when the pump is operating at its design conditions and at the most limiting operating conditions. Discuss the size of particles that could pass through the strainer into the RHR pump passages. Indicate the amount of material blockage it would take to significantly reduce the RHR pump suction flow from the suppression pool following a postulated LOCA.

Response:

The suction strainers are currently being purchased to new criteria as defined by the Mark II Hydrodynamic Load Program. The specifications for the strainer were provided in the response to Question 022.039.* The location of the suction strainers inside the suppression pool, as well as the pipe bends, are shown on Figures 211.023-1 and 211.003-2. Outside the suppression pool, RHR A and B loops have five elbows and RHR C has six elbows.

The minimum height of the suppression pool water level is el. 466'-0 3/4" and is controlled by the technical specifications. The centerline elevation of all the RHR suction strainers is el. 447'-0". The minimum NPSHA is calculated per Regulatory Guide 1.1. The calculation for RHR pumps is outlined in the response to Question 022.038. The friction loss in the suction piping for all the RHR loops is approximately 3 feet at 7900 gpm. Using the very conservative assumptions outlined in the response to Question 022.038, the minimum NPSHA is 36 ft., while the maximum required NPSH is about 13 ft. at 7900 gpm. (Reference Figures 6.3-10a, 6.3-10b, and 6.3-10c, RHR pump performance curves). The new suction strainers being purchased to meet the Mark II Hydrodynamic Loads will increase the suction friction losses, but with 23 feet of margin between the minimum NPSHA and maximum NPSHR at the most limiting operating condition, this will not affect the safe operation of the RHR pumps. Friction loss through the new strainer is expected to be about 4 feet with 50% of the surface clogged. This will be verified during preoperational testing.

The pump manufacturer imposed a maximum particle size of three thirty-seconds (3/32) of an 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 containment spray nozzles which have an orifice opening of seventeen sixty-fourths (17/64) of an inch. Accordingly, the strainers are specified to prevent the passage of particles three-thirty seconds of an inch or greater. Since the water in the suppression pool is kept at a high quality, clogging of the strainers is not considered credible. Reflective insulation panels are used exclusively within the primary containment and constitute the only credible debris within the primary containment following a LOCA or seismic event. Blockage of the ECCS strainers by this debris is not considered credible. However, the RHR system is designed to have adequate NPSHA with the suction strainers 50% clogged as noted in the response to Question 022.038.

*Advance copy of response was submitted on WPPSS to NRC letter (Renberger to Rubenstein), G02-79-144, dated August 14, 1979.

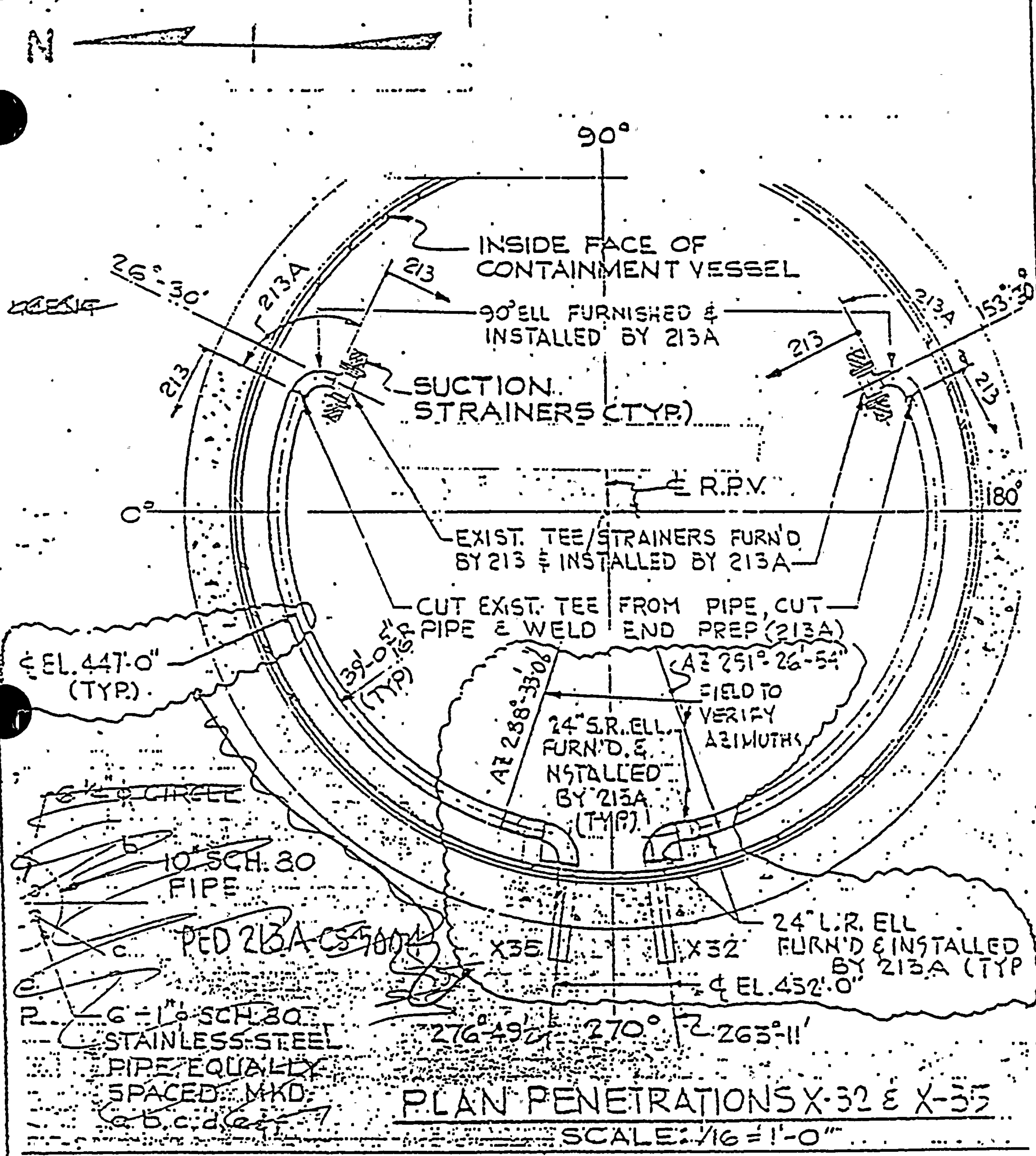
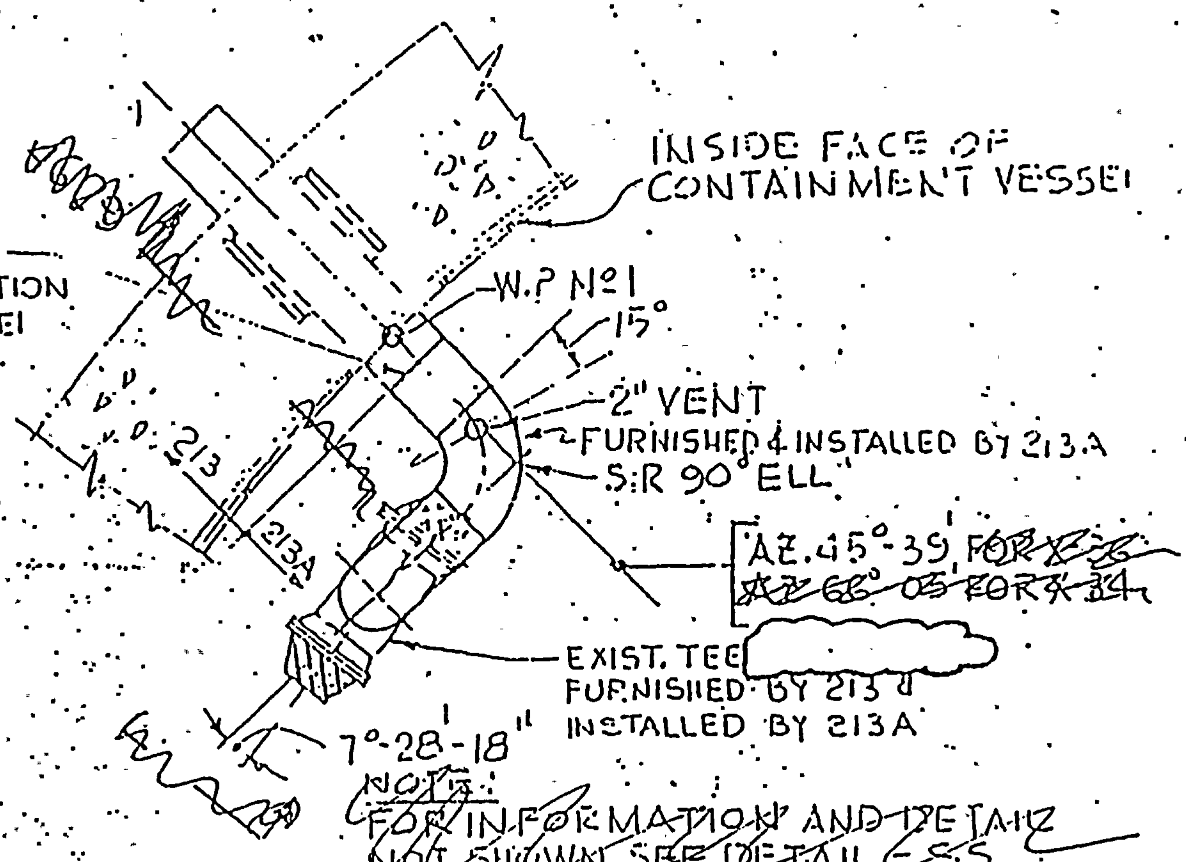


FIGURE 211.33-1
RHR A AND B SUCTION LINES
INSIDE THE SUPPRESSION POOL

REF. DOC./PCN 6650	RFI.	- WPPSS NUCLEAR PROJECT NO. 2	
REF. SPEC. SECTION: 213A PAGE: 15A-27 PARA:		BURNS AND ROE, INC.	
REF. DWG.: 5798 REV. 27	DWG. ZONE G, H - 1, 2, 3	PED 213A-CS-5004	SHT 3 OF 5
SCALE: DRAWN BY J.F. DATE 3/7/79	REVIEWED DATE 3/7/79	TITLE R.B	



PLAN PENETRATION X-36
SCALE: 3/16" = 1'-0"

FIGURE 211.33-2
RHR C SUCTION LINE
INSIDE THE SUPPRESSION POOL

REF. DOC./PCN: 6433	RFI
REF. SPEC. SECTION: ——	PAGE: —— PARA: ——
REF. DWG. 5798 REV. 27	DWG. ZONE: E1
SCALE 3"=1'-0"	FED/213A-CS-0101 SHT 5 OF 6
DRAWN BY FROEP DATE 2/27/79 REVIEWED DATE 7/27/79	BURNS AND ROE, INC.
APPVD: DATE:	NUCLEAR PROJECT NO. 2
TITLE: CONTAINMENT VESSEL	

Q. 211.034
(5.4.7)

Provide the process data (i.e., flow, temperature, and pressure) for all modes of operation of the RHR system which you reference in Figure 5.4-14a of the FSAR (i.e., MPL Item No. E12-1020).

Response:

Figures 5.4-14b and 5.4.14c provide all the process data for all modes of operation of the RHR system. Figure 5.4-14b was inadvertently left out of the FSAR, but was included in Amendment 3, March, 1979, as part of the response to FSAR Question 212.004.

Q. 211.035
(5.4.7)

Identify the pressure interlock setpoints of the RHR isolation valves F008 and F009 which are set to: (1) prevent inadvertent openina to the low oressure suction oioina, and (2) initiate valve closure when the reactor pressure is increasing.

Response:

The pressure interlock setpoint for RHR shutdown suction isolation valves F008 and F009 is nominally 135 psig. The setpoint for opening and closing is the same.

Q. 211.036
(5.4.7)

Confirm that all valves performing an isolation function between the high pressure and low pressure boundary in the RHR system (e.g., check valves and motor-operated valves) meet the leak testing and inspection requirements of Section XI of the ASME code for Category A valves. In this regard, it is our position that a combination of two or more check or motor-operated valves in series should have design provisions which permit individual leak testing of any two valves.

Response:

All valves performing an isolation function between the high pressure and low pressure boundary in the RHR system meet the leak testing and inspection requirements of Section XI of the ASME code for Category A valves. The RHR connects to high pressure systems for RPV head spray, shutdown cooling suction, low pressure coolant injection, and shutdown cooling return. Design provisions for these valves to permit individual leak testing of any two valves performing an isolation function is shown in the following table:

Penetration No.	Function	Isolation Valves	Design Provisions
2	RPV Head Spray	RHR-V-23 RCIC-V-66	Fig. 6.2-31e
20	Shutdown Cooling Suction	RHR-V-8 RHR-V-9	Fig. 6.2-31k
12A, B, C	Low Pressure Coolant Injection	RHR-V-42A, B, C RHR-V-41A, B, C	Fig. 6.2-31l
19A, B	Shutdown Cooling Return	RHR-V-53A, B RHR-V-50A, B RHR-V-123A, B	Fig. 6.2-31m

Q. 211.037
(5.4.7)

In Note 12 of Figure 5.14-13a of the FSAR, you state that: "Between valves MOF008 and MOF009 consideration should be given to thermal expansion of the contained water." Provide a commitment to incorporate a method for pressure relief between these two RHR isolation valves. Alternatively, show by analysis that piping integrity would be maintained in the event that a LOCA or stream line break occurred and the water trapped between these two valves, thermally expanded.

Response:

A pressure relief valve will be installed between MOF008 and MOF009. It will meet the following design requirements:

ASME III, Class 1	
Seismic Category 1	
Quality Class 1	
Relief Capacity	10 gpm
Design Pressure	1250 psig
Design Temperature	575°F
Set Pressure	150 psig
Size	3/4" x 1" flanged

The discharge of the relief valve will be back to the equipment drains.

Q. 211.038
(5.4.7)
(14.2.12)

Provide the test acceptance criteria discussed in Section 14.2.12.1.7 of the FSAR regarding preoperational testing of the RHR system.

Response:

Tentative acceptance criteria for the RHR preoperational test is attached.

Acceptance Criteria for RHR Preoperational Test

1.	RHR-V-42A, B, C	LPCI injection valve maximum opening time: 27 seconds
2.	RHR-V-8 RHR-V-9	Shutdown cooling suction valve maximum closing time: 40 seconds
3.	RHR-V-53A, B	Shutdown cooling return valve maximum closing time: 40 seconds
4.	RHR-FCV-64A, B, C	Minimum flow valve maximum opening/closing time: 15 seconds.

5. Motor operated valves open with initial differential pressure of:

RHR-V-42A, B, C	Loop A injection	750 psi
RHR-V-16A, B	Containment Spray	SOH (shutoff head)
RHR-V-17A, B	Containment Spray	SOH
RHR-V-27A, B	Suppression Pool Spray	SOH
RHR-V-24A, B	Suppression Pool Test Return	SOH
RHR-V-21	Suppression Pool Test Return	SOH
RHR-V-53A, B	Shutdown Cooling Return	SOH
RHR-V-8	Shutdown Cooling Suction	150 psi
RHR-V-9	Shutdown Cooling Suction	150 psi
RHR-V-23	Reactor Vessel Head Spray	3OH
RHR-V-4A, B, C	LPCI Suppression Pool Suction	70 psi
RHR-V-64A, B, C	Minimum Flow	500 psi
RHR-V-123A, B	Shutdown Cooling Check Bypass	1050 psi

6. RHR pump flow at 26 psi pressure differential between the RPV suppression pool: 7450 gpm minimum.

7. RHR System flow with three (3) pumps at 26 psi pressure differential between RPV suppression pool: 22,350 gpm minimum.

8. Low pressure coolant injection initiation to rated flow with normal auxiliary power available: 27 seconds maximum.

9. Low pressure coolant injection to rated flow from time diesel generator initiation from loss of auxiliary power: 37 seconds maximum.

10. NPSHA in mode B, post-accident containment spray, 7900 gpm and 220°F suppression pool suction with 50% strainer flow: (Later) ft.

11. NPSHA in mode D, shutdown cooling, flow 7450 gpm and 335°F suction temperature: (Later) ft.

Q. 211.039
(5.4.7)

Operation of the RHR system in the steam condensing mode involves partial draining of one or both RHR heat exchangers and introduction of reactor steam into lines and heat exchangers which are initially cold. Describe the methods (e.g., valve operation or air introduction) and the provisions you propose to prevent the occurrence of water hammer during initiation of operation in this mode and in the change to the pool cooling mode. Indicate whether the jockey pump system shown in Figure 5.4-13a of the FSAR can fill the lines to the injection valve in the core spray lines and the RHR lines (i.e., valves F016 and F042, respectively) when the RHR is in the steam condensing mode using one or both heat exchangers. If not, indicate what procedures you propose to prevent water hammer following startup of the core spray or RHR pumps.

Response:

Refer to Figure 5.4-13a for valve numbers. The methods used to prevent the occurrence of water hammer during steam condensing initiation are:

(1) lowering the heat exchanger water level using low pressure steam (approximately 10 psig) by cracking open steam pressure control valve bypass valve F087;

(2) initially admitting steam at a low pressure and slowly increasing steam pressure to 200 psig to avoid high pressure surges; and

(3) opening all valves slowly to avoid sudden flow surges.

The methods used to prevent the occurrence of water hammer following steam condensing termination and change to the pool cooling mode are:

(1) closing the heat exchanger condensate discharge;

(2) opening the valves connecting the heat exchanger to the main pump loop (F003 and F047); and

(3) opening the high point vent and filling the heat exchanger shell and connecting piping using the condensate supply valve.

When the RHR system is used for steam condensing, the LPCI injection loop is isolated from the heat exchanger steam flow by closing F003 and F047. Use of steam condensing mode has no effect on the jockey pumps' ability to fill the lines to the injection valves in the core spray or RHR lines because the heat exchanger bypass valve F048 is open. Therefore, the jockey pumps can fill these lines.

Q. 211.040
(5.4.7)

Those pressure relief valves and lines which are designed to prevent overpressurization of the RHR system, are routed outside the containment before being returned to the suppression pool. Discuss the design provisions incorporated into the WNP-2 facility to minimize the potential for water hammer in these lines. State whether these relief lines are capable of withstanding both seismic and dynamic blowdown loads without suffering a loss of structural integrity.

Response:

Except where noted below, the RHR relief valves are installed to accommodate thermal expansion and leakage across closed valves in isolated piping systems (see response to Question 211.027 for additional information on RHR relief valves). Pressure buildups in isolated lines will be slow and discharges from the relief valves in these lines will be small. Water hammer and other hydrodynamic loads are not considered a potential problem in those lines.

RHR-RV-55A and B and RHR-RV-95A and B (reference Figure 3.2-6, zones E, 4 and E, 14) are steam relief valves which protect the RHR heat exchangers from overpressure in case RHR-PCV-51A and B fail during the RHR steam condensing mode. There is no potential for water hammer in the discharge line of RHR-RV-95A and B, which have their own discharge line into the suppression pool. Since the discharge lines for RHR-RV-55A and for RHR-RV-55B share a common pipe with several other RHR lines which could fill the discharge lines with water during other modes of RHR operation, e.g., system test, an automatic vacuum breaker is being added to ensure that the water level in these discharge lines is at the suppression pool water level during the steam condensing mode.

In addition, these steam relief valves have an automatic drain pot to prevent any water from accumulating ahead of the valves.

RHR-RV-36 (Figure 3.2-6, zone G, 13) is a water relief valve which protects the lower pressure rated PCIC suction piping in case of either or both RHR-LCV-65A and RHR-LCV-65B failing open during the steam condensing mode. The discharge line for RHR-RV-36 uses the same pipe as RHR-RV-55A, where an automatic vacuum breaker guarantees that there is no water in the pipe.

It should be noted that the probability of the RHR steam relief valves or RHR-RV-36 actuating is extremely low. These relief valves can actuate only during the RHR steam condensing mode which is expected to be used only eight hours per year. In addition, RHR-PCV-51A and B and RHR-LCV-65A and B are designed to fail closed.

RHR relief lines (identified by their value tag numbers RHR-RV-36, RHR-RV-55A, RHR-RV-55B, RHR-RV-95A, and RHR-RV-95B) are capable of withstanding both seismic and dynamic blowdown loads without suffering a loss of structural integrity.

Q. 211.041
(5.4.7)

Discuss the procedures to be used in the WNP-2 facility which will minimize the potential for exceeding the allowable cooldown rate (i.e., a cooldown rate greater than 100 degrees Fahrenheit/hour) of the RHR system and the reactor coolant system when placing the WNP-2 facility in a shutdown cooling mode following normal shutdown or following an emergency shutdown.

Response:

The potential of exceeding the 100°F/hr cooldown limit during the cooldown mode will be minimized by the following elements of the operating procedures:

(1) Only one RHR loop is initially put into cooldown mode operation. The second loop will not be put into cooldown operation until the full cooling capacity of the one loop is inadequate to maintain the desired cooldown rate.

(2) The RHR loop is started with the heat exchanger bypass full open. This limits the flow through the heat exchanger to about 40 to 45% of the total loop flow.

(3) The cooldown injection valve E12-F053 is throttled to estalbish the desired total loop flow rate.

(4) As stable operation is established and the initial cooldown rate defined the flow through the RHR heat exchanger will be adjusted with the bypass valve E12-F048 or the cooldown injection valve E12-F053 to adjust the cooldown rate to the deisred level.

The minimum flow through the heat exchanger will be required at the start of the cooldown mode. As the reactor is cooled the flow must be increased in order to maintain a given cooldown rate. Thus, once an acceptable cooldown rate is established it will not be exceeded.

Q. 211.042
(5.4.7)

Discuss the reliability of the RHR pumps for long-term operation. It is our position that long-term reliability of these pumps should be demonstrated either by operational experience or by testing. If you cite previous operational experience as the basis for qualifying these pumps in your response to this question, identify any design differences in the pumps and indicate the operating conditions of the pump service life which is cited.

Response:

The RHR pumps are designed for the life of the plant (40 years) and tested for operability assurance and performance as follows:

a. In-shop tests, including (1) hydrostatic tests pressure retaining parts of 150% times the design pressure, (2) performance tests while the pump is operated with flow to determine the total developed head at zero flow and design flow, and (3) net positive suction head (NPSH) requirements.

b. After the pump is installed in the plant, it undergoes (1) the system hydro test, (2) functional tests, (3) the required periodic inservice inspection and operation of once a month, and (4) about one month of operation each year for a refueling shutdown.

c. In addition, the pumps are designed for a postulated single operation of 3 to 6 months for one accident during the unit's 40 year life.

A listing of GE operating experience of Ingersoll-Rand RHR pumps is provided in the response to Question 211.072.

Q. 211.043
(5.4.6)

Provide an RCIC pump performance curve that shows flow rate versus reactor vessel pressure. Identify the most limiting operating condition for the RCIC pump and identify the NPSH margin under this condition.

Response:

RCIC system is designed to provide vessel makeup flow of 600 gpm for varying reactor vessel pressures from 150 psig to 1158 psig. Consequently, there is no RCIC pump performance curve that depicts flow rate versus reactor vessel pressure. Two performance curves are provided; one for constant flow and the other for constant speed. These figures will be added to Section 5.4. as Figures 5.4-19a and 5.4-19b, respectively.*

The available NPSH is calculated for worst-case conditions (i.e., 600 gpm rated fluid flow, maximum fluid temperature), which is RCIC suction from the suppression pool. Using the conservative water temperature of 170°F, the minimum NPSHA is 56 ft. For the case where RCIC is pulling suction from the condensate storage tank and for the water temperature of 100°F, there is 48 ft. of NPSHA at 600 gpm. At the high speed set point for the RCIC turbine (4500 rpm), the required NPSH at 600 gpm is 19 ft. Therefore, there is adequate NPSH margin whether the RCIC system is pulling suction from the suppression pool or the condensate storage tank.

*Draft FSAR page changes attached.

5.4.6.2.2.2 Design Parameters

Design parameter for the RCIC system components are listed below. See Figure 5.4-9 for cross-reference of component numbers listed below:

a. RCIC Pump Operation (C001) (REFER TO FIGURES 5.4-19a AND 5.4-19b)

Flow rate	Injection Flow - 600 gpm Cooling Water Flow - 16-25 gpm Total Pump Discharge - 625 gpm (Includes no margin for Pump Wear)
Water Temperature Range	$40^{o}F$ to $140^{o}F$
NPSH	21 ft minimum
Developed Head	2890 ft @ 1155 psia Reactor Pressure 610 ft @ 165 psia Reactor Pressure
BHP, Not to Exceed	725HP @ 2890 feet Developed Head 130HP @ 610 feet Developed Head
Design Pressure	1500 psia
Design Temperature	$40^{o}F$ to $140^{o}F$

b. RCIC Turbine Operation (C001)

	H.P. Condition	L.P. Condition
Reactor Press (Sat. Temp.)	1155 psia	165 psia
Steam Inlet Pressure	1140 psia	150 psia
Turbine Exhaust Press	15 to 25 psia	15 to 25 psia
Design Inlet Pressure	1250 psia + saturated temperature	
Design Exhaust Pressure	165 psia + saturated temperature	

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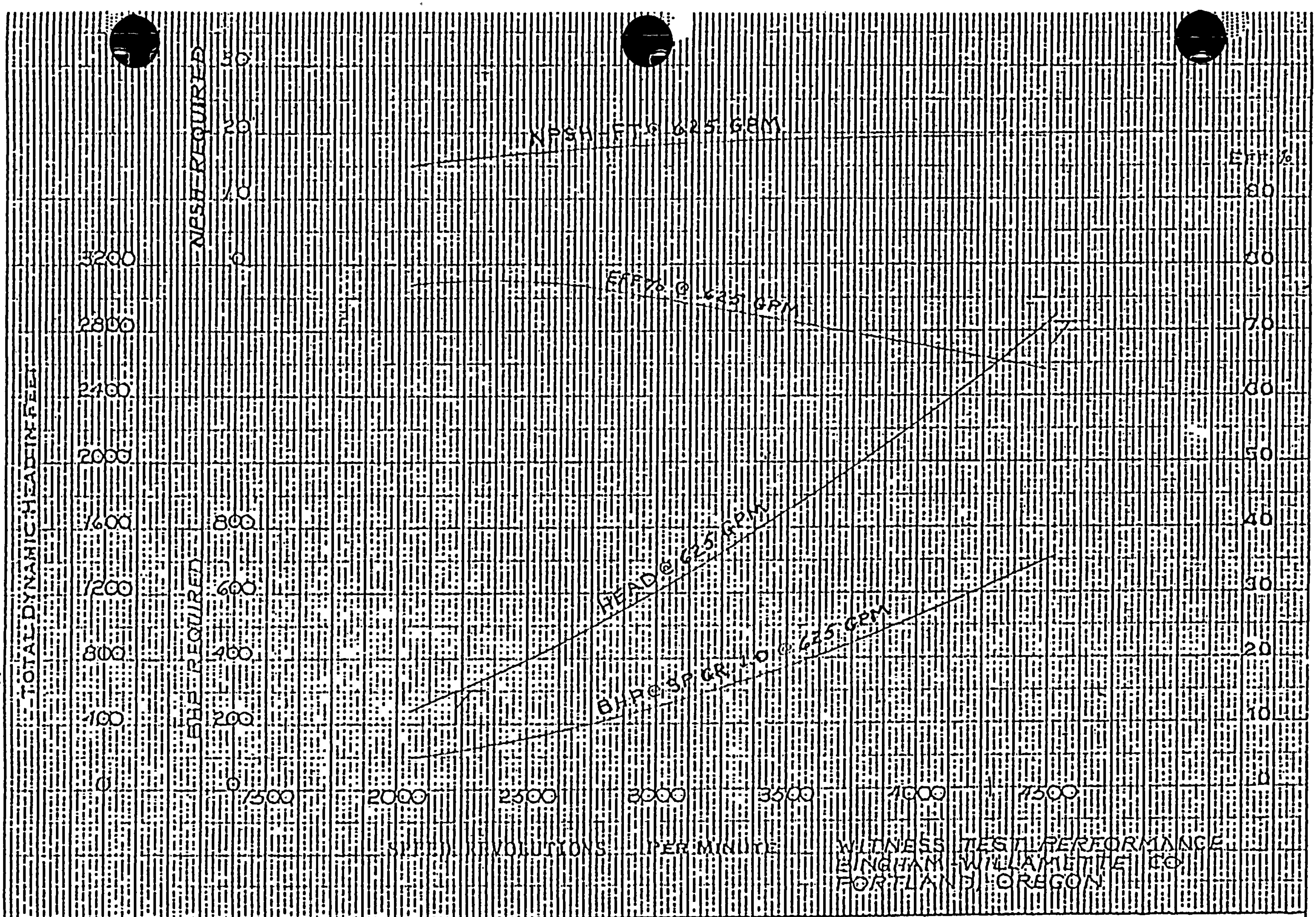
LIST OF FIGURES (Continued)

C. [illegible] 9/10/79



GENERAL ELECTRIC CO.

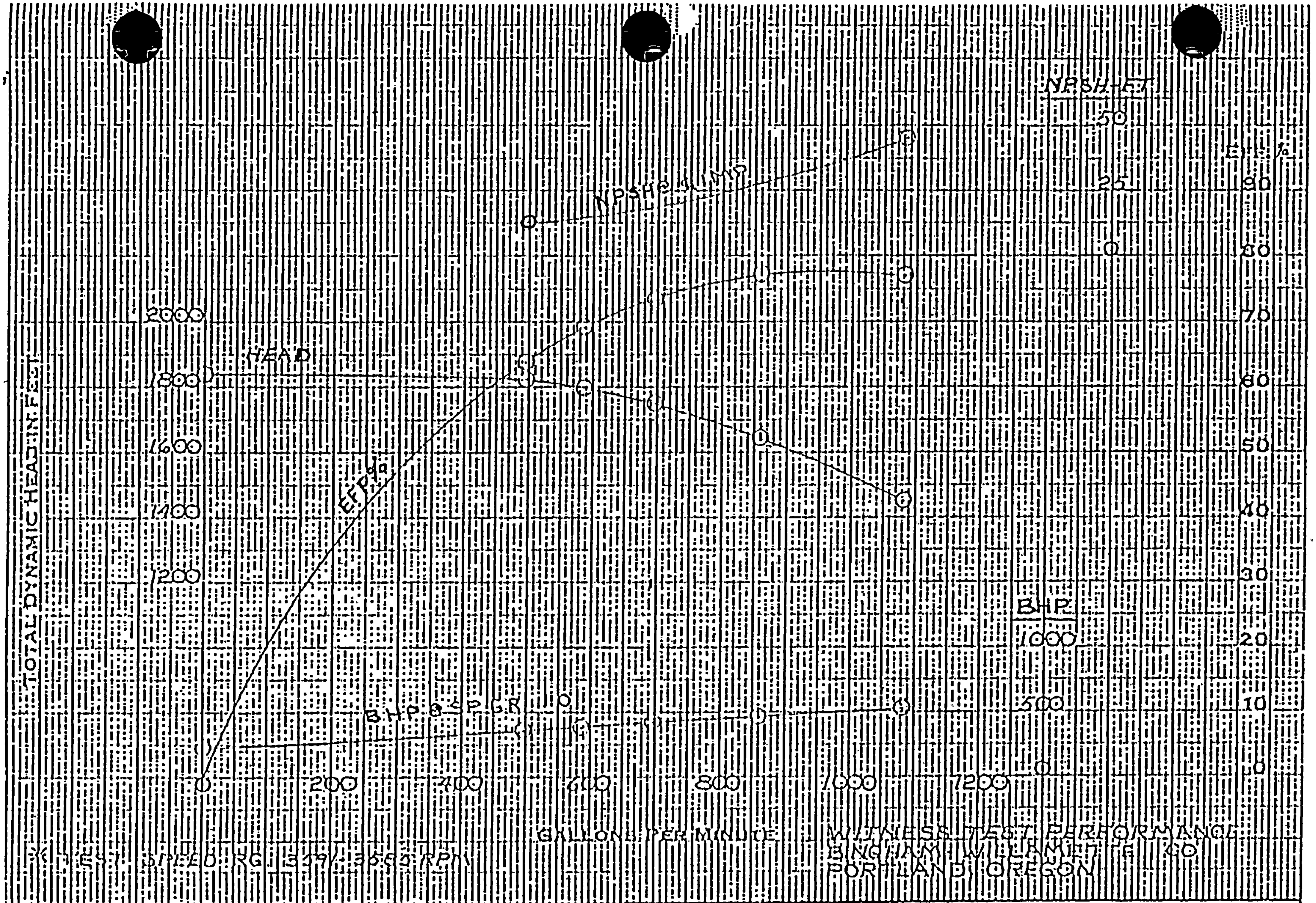
PUMP No: 210074

CHARACTERISTIC CURVE SHEET
PUMP ENGINEERING DEPT.
BINGHAM-WILLAMETTE COMPANY
PORTLAND OREGON & SHREVEPORT LA
. C.H. 3-7-73

IMPELLER MAX. DIA.	10 3/4	6 X 6 X 10 1/2 CP 4 STG PUMP		
MIN. DIA.		DIA. IMPELLER 10 5/16	IMPELLER PATT. 1) 413MSD-11 2-4) 413MSD-2	625 G.P.M.
EYE AREA 19.5 SQ. IN.		N.P.S.H. REQUIRED	REFERENCE 31705	CURVE NO. 31706

Form No. 312

FIGURE 5.4-19a RCIC PUMP PERFORMANCE CURVE (CONSTANT FLOW)



GENERAL ELECTRIC CO.	CHARACTERISTIC CURVE SHEET	IMPELLER MAX. DIA. 10 3/4	6 X 6 X 10 1/2 CP 4 STG PUMP	
	PUMP ENGINEERING DEPT.	MIN. DIA.	DIA. IMPELLER 10 5/16"	IMPELLER PATT. 1) 413MSD-11 2-4) 413MSD-2 / 3585* R.P.M.
PUMP No 210074	BINGHAM-WILLAMETTE COMPANY PORTLAND, OREGON & SHREVEPORT, LA C.H. 3-7-73	EYE AREA 19.5 SQ. IN.	N.P.S.H. REQUIRED	REFERENCE 31706 / CURVE No. 31705

Form No. 312

FIGURE 5.4-19b RCIC PUMP PERFORMANCE CURVE

Q. 211.044
(5.4.6)

When the steam isolation valves are temporarily closed for maintenance, it appears that it is possible for some steam condensate to remain in the lines leading to the RCIC steam turbine. Discuss whether the amount of water condensed from steam can cause sufficient damage to the RCIC turbine to render the RCIC system incapable of delivering water to the reactor vessel as required. Describe the design modifications, if any, you propose to prevent water hammer occurring at the RCIC turbine exhaust.

Response:

Refer to Figure 5.4-9a, RCIC P&ID, for valve numbers. If the steam isolation valves were temporarily closed for maintenance, administrative control and specific operating procedures relieve the possibility of thermal shock or water hammer to the steamline, valve seals and discs. Keylock switches are provided for positive administrative control. Operating procedures require throttling open the outboard isolation valve, F008 to remove any condensate trapped between the isolation valves warming up the steamline by throttling open the warmup valve F076 located on a pipeline bypassing the inboard isolation valve, and then opening the inboard isolation valve F063. All the condensate is removed from the steam supply line by a drain pot located at the lowest point.

A vacuum breaker system is installed close to the RCIC turbine exhaust line suppression pool penetration to avoid siphoning water from the suppression pool into the exhaust line, as steam in the line condenses during and after turbine operation. The vacuum breaker line runs from the suppression pool air volume to the RCIC exhaust line through two normally open motor-operated gate valves F080 and F086 and two swing check valves arranged to allow air flow into the exhaust line and to preclude steam flow to the suppression pool air volume. Condensate buildup in the turbine exhaust line is removed by a drain pot in the low point of the line near the turbine exhaust connection. The condensate collected in the drain pot drains to the barometric condenser.

Q. 211.045
(5.4.6)

An isolation signal will close a number of valves (i.e., F063 and F008) in the RCIC system, located inside and outside containment, branched off the main steamline. However, the process and instrumentation drawing (P&ID) for the RCIC (i.e., Figure 5.4-9a) shows that these valves are keylocked open. Explain this apparent discrepancy. Additionally, evaluate the consequences of a postulated pipe break downstream of the first or second isolation valve for steam flow rates which are greater than 300% of the steady-state steam flow. Provide justification for the selection of this 300% limit.

Response:

The keylocked switches for F063 and F008 do not prevent automatic isolation of these valves. The keylocked switches are provided to prevent inadvertent manual isolation of the RCIC steam supply during normal operation. These valves are normally left in the open position. An isolation signal is given for a large pipe break by detecting flow rates greater than 300% of the steady-state steam flow. For leakage with flow rates less than 300% of steady-state steam flow, an isolation signal is signaled by use of area temperature sensors provided by the leak detection system.

If the steam isolation valves were temporarily closed for maintenance, operating procedures give specific directions on opening the steam isolation valves and the warmup line. This administrative control relieves the possibility of thermal shock or water hammer to the steamline, valve seats and discs. Keylock switches on the steam isolation valves provide positive administrative control of the operating procedures.

RCIC isolation for steam flow rates which are greater than 300% of the steady-state steam flow is a sufficiently high setpoint to avoid inadvertent isolation due to startup transients and is high enough to detect large pipe breaks. Small pipe breaks are detected by the leak detection system.

Q. 211.046
(5.4.6)

The acceptance criteria contained in Section 5.4.6 of the SRP state '"As a system which must respond to certain abnormal events, the RCIC system must be designed to sesimic Category I standard, as defined in Regulatory Guide 1.29". However, the condensate storage tank, which is the normal supply of water for the RCIC, is not designed to seismic Category I criteria. While the suppression pool provides an alternate source of water from a seismic Category I structure, the switchover to this alternate source in the WNP-2 facility requires operator action. Any one of the following alternatives would be an acceptable approach for meeting the acceptance criterion cited above: (1) a seismic Category I supply; (2) an automatic safety-grade switchover to a seismic Category I supply; or (3) a manual safety-grade switchover to a seismic Category I supply, if appropriately justified. It appears that you are proposing to use the third option. If so, provide justification for the time required for the operator to perform a manual switchover. If the third alternative is not proposed, identify and discuss which approach will be used in the WNP-2 facility.

Response:

As stated in our response to Question 031.015, WNP-2 is providing an automatic safety-grade switchover to a seismic Category I supply (the suppression pool), which satisfies your alternative (2).*

*Draft FSAR page changes attached.

The suction line from this source is provided with an in-line reserve tank with appropriate safety-related level instrumentation. In the event that the water supply from the condensate storage tank becomes exhausted, the level instrumentation in the in-line reserve tank initiates an automatic switchover to the suppression pool as the water source for the RCIC pump. The in-line reserve tank has sufficient volume to maintain the minimum required RCIC pump NPSH plus a two foot margin while the switchover occurs, thus assuring a water supply for continuous operation of the RCIC system.

where the shutdown coolant system can be placed into operation.

Following a reactor scram, steam generation will continue at a reduced rate due to the core fission product decay heat. At this time the turbine bypass system will divert the steam to the main condenser, and the feedwater system will supply the make-up water required to maintain reactor vessel inventory.

In the event the reactor vessel is isolated, and the feedwater supply is unavailable, relief valves are provided to automatically (or remote manually) maintain vessel pressure within desirable limits. The water level in the reactor vessel will drop due to continued steam generation by decay heat.

Upon reaching a predetermined low level, the RCIC System is initiated automatically. The turbine driven pump will supply demineralized make-up water from the condensate storage tank to the reactor vessel. ~~an alternate source of water is available from the suppression pool.~~ The turbine will be driven with a portion of the decay heat steam from the reactor vessel, and will exhaust to the suppression pool.

During RCIC operation, the suppression pool shall act as the heat sink for steam generated by reactor decay heat. This will result in a rise in pool water temperature. Heat exchangers in the Residual Heat Removal System are used to maintain pool water temperature within acceptable limits by cooling the pool water directly or by condensing generated steam prior to entering the suppression pool. When using the steam condensing mode, the condensate discharge from the heat exchangers may be used as RCIC pump suction supply.

5.4.6.2.1.2 Diagrams

The following diagrams are included for the RCIC Systems.

a. A schematic "Piping and Instrumentation Diagram" (Figure 5.4-9) shows all components, piping, points where interface system and subsystems tie together and instrumentation and controls associated with subsystem and component actuation.

b. A schematic "Process Diagram" (Figure 5.4-10) shows temperature, pressures, and flows for RCIC operation and system process data hydraulic requirements.

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OS-4

1 of 3

Steam Supply Isolation Valves (F008)	Open and/or close against full differential pressure of 1140 psi at a minimum rate of 12 inches per minute.
RHR Steam Supply Isolation Valves (F063/F064)	Open and/or close against full differential pressure of 1140 psi.
Cooling Water Pressure Control Valve (F015)	Self-contained downstream sensing control valve capable of maintaining constant downstream pressure of 75 psia.
Pump Suction Relief Valve (F017)	100 psig relief setting; 10 gpm at 10 percent accumulation.
Cooling Water Relief Valve (F018)	Sized to prevent over pressurizing piping, valves and equipment in the coolant loop in the event of failure of pressure control valve F015.
Pump Test Return Valve (F022)	Is capable of throttling against 1000 psi differential pressure.
Relief Valve Barometric Condenser (F033)	Relief valve is capable of retaining 10 inches of mercury vacuum at 140°F ambient, with a set pressure of 5-7 psig and a flow of 20 gpm at 10 percent accumulation.
Pump Suction Valve Suppression Pool (F031)	Is located as close as practical to the primary containment.
Pump Suction Valve Condensate Storage Tank (F010)	Open and/or close against full differential pressure of 45 psi within 15 seconds.

l. High and low inlet RCIC steam line drain pot levels, respectively, open and close F054.

m. The combined signal of low flow plus discharge pressure open and with increased flow closes F019. Also see items e and f above.

n. The signal of in-line reserve tank low water level opens valve F031.

5.4.6.2.2 Equipment and Component Description

5.4.6.2.2.1 Design Conditions

Operating parameters for the components of the RCIC Systems, defined below, are shown on Figure 5.4-10.

a. One 100% capacity turbine and accessories

b. One 100% capacity pump assembly and accessories

c. Piping, valves, and instrumentation for:

1. Steam supply to the turbine

2. Steam supply to RHR condensing heat exchanger

3. Turbine exhaust to the suppression pool

4. Make-up supply from the condensate storage tank to the pump suction

5. Make-up supply from the suppression pool to the pump suction

6. Make-up supply from the RHR steam condensing heat exchangers

7. Pump discharge to the head cooling spray nozzle, including a test line to the condensate storage tank, a minimum flow bypass line to the suppression pool, and a coolant water supply to accessory equipment.

The basis for the design conditions was the American Society of Mechanical Engineery (ASME) Section III, Nuclear Power Plant Components.

storage tank falls below a predetermined level, ~~as indicated by an annunciator~~, the suppression pool suction valve is AUTOMATICALLY opened~~, by a manual switch~~. When the suppression pool suction valve is open~~ed~~, the condensate storage tank suction valve automatically closes.

One d-c motor-operated RCIC pump discharge valve in the pump discharge pipeline is provided. The control scheme for this valve is shown in Figure 7.4-2. This valve is arranged to open upon receipt of the RCIC initiation signal and to close automatically upon receipt of a turbine trip signal.

7.4.1.1.3.7 Separation

As in the emergency core cooling system, the RCIC system is separated into divisions designated 1 and 2. The RCIC is a Division 1 system, but the inboard steam line isolation valve, the steam line warmup line isolation valve, and the inboard vacuum breaker isolation valve are in Division 2; therefore, part of the RCIC logic is Division 2. The inboard steam supply line isolation valve and the steam line warmup line isolation valve are a-c powered valves. The rest of the valves are d-c powered valves. In order to maintain the required separation, RCIC logic relays, instruments and manual controls are mounted so that separation from Division 2 is maintained.

All power and signal cables and cable trays are clearly identified by division and safety classification. The method used for identifying power and signal cables and cable trays and the method of identifying non-safety related cables as associated circuits are discussed in 8.3.1.3.

7.4.1.1.3.8 Testability

The RCIC may be tested to design flow during normal plant operation as discussed in 7.4.1.1.3.1. Water is drawn from the condensate storage tank and discharged through a full flow test return line to the condensate storage tank. The discharge valve from the pump to the feedwater line remains closed during the test and reactor operation remains undisturbed. Design of the control system is such that the RCIC system returns to the operating mode from test if system initiation is required.

Testing of the initiation transducers which are located outside the drywell is accomplished by valving out each transducer and applying a test pressure source. This verifies the operability of the sensor as well as the calibration range. In control room observations, observation of relay contact closure of the relays directly coupled to the initiation transducers verify the operability of the instrument channel.

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Q. 211.047
(15.2.9)

Provide the suppression pool temperature and the reactor vessel temperature and pressure as a function of time for the alternate shutdown cooling modes (i.e., activity C1 and C2) described in Figure 15.2-11 of the FSAR, assuming a failure of the normal RHR shutdown cooling mode. Provide an estimate of the time required to achieve a cold shutdown condition for these alternate cooling paths. Identify the initial pool and service water temperatures assumed in this analysis.

Response:

The suppression pool temperature and the reactor vessel temperature and pressure response for the alternative shutdown cooling modes is shown on new Figures 15.2-16, 15.2-17, 15.2-18 and 15.2-19. Cold shutdown is achieved in approximately 36 hours for activity C1.6.2 or in approximately 15 hours to activity C1.6.1 or C2. The initial pool and service water temperatures assumed in this analysis are 95°F and 87°F, respectively, as shown in new FSAR Table 15.3-13.*

*See attached draft page changes.

15.2.9.4 Qualitative Results

For most single failures that could result ~~is~~ in loss of shutdown cooling, no unique safety actions are required. In these cases, shutdown cooling is simply re-established using the redundant shutdown cooling loop. In cases where the RHR shutdown cooling suction line valves cannot be opened, alternate paths are available to accomplish the shutdown cooling function (Figure 15.2-10). An evaluation has been performed assuming a failure that disables the RHR shutdown cooling suction line valves.

This evaluation demonstrates the capability to safely transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

The alternate cooldown path chosen to accomplish the shutdown cooling function utilizes the RHR and ADS or normal relief valve systems (see Reference 15.2-3 and Figure 15.2-11). The alternate shutdown systems are capable of performing the function of transferring heat from the reactor to the environment using only safety systems. The systems are capable of bringing the reactor to a cold shutdown in APPROXIMATELY ~~less than~~ ~~8~~ 36 hours OR LESS after the transient occurs.

The systems have suitable redundancy in components such that even for onsite electrical power operation (offsite power is not available) the systems' safety function can be accomplished assuming an additional single failure. The systems can be fully operated from the main control room.

The design evaluation is divided into two phases: (1) full power operation to approximately 100 psig vessel pressure, and (2) approximately 100 psig vessel pressure to cold shutdown (14.7 ~~psia~~ 200°F) conditions.

15.2.9.4.1 Full Power to Approximately 100 psig

Independent of the event that initiated plant shutdown (whether it be a normal plant shutdown or a forced plant shutdown), the reactor is normally brought to approximately 100 psig using either the main condenser or, in the case where the main condenser is unavailable, the RCIC/HPCS systems together with the nuclear boiler pressure relief system and the RHR heat exchanger in the suppression pool cooling mode.

(LOSS OF OFFSITE POWER)

For evaluation purposes, however, it is assumed that plant shutdown is initiated by a transient event which results in relief valve actuation and subsequent suppression pool heatup. For this postulated condition, the reactor is shutdown and the reactor vessel pressure is reduced to approximately 100 psig. Manual operation of the safety/relief valves is utilized to ~~control pressure~~. Reactor vessel makeup water is automatically provided via the RCIC/HPCS systems. While in this condition, the RHR system (suppression pool cooling mode) is used to maintain the suppression pool temperature within shutdown limits.

DEPRESSURIZE THE REACTOR VESSEL

These systems are designed to routinely perform their functions for both normal and forced plant shutdown. Since the RCIC/HPCS and RHR systems are divisionally separated, no single failure, together with the loss of offsite power, is capable of preventing reaching the 100 psig level.

15.2.9.4.2 Approximately 100 psig to Cold Shutdown

The following assumptions are used for the analyses of the procedures for attaining cold shutdown from a pressure of approximately 100 psig:

a. The vessel is at 100 psig and saturated conditions.

b. A worst-case single failure is assumed to occur (i.e., loss of a division of emergency power) and

c. There is no offsite power available.

In the event that the RHR's shutdown suction line is not available because of single failure, the first action to be taken will be to maintain the 100 psig level while personnel gain access and effect repairs. For example, if a single electrical failure caused the suction valve to fail in the closed position, a hand wheel is provided on the valve to allow manual operation. Nevertheless, if for some reason the normal shutdown cooling suction line cannot be restored to service, the capabilities described below will satisfy the normal shutdown cooling requirements and thus fully comply with GDC 34.

TABLE 15.2-12

SEQUENCE OF EVENTS FOR FAILURE OF RHR SHUTDOWN COOLING

Approximate Elapsed Time	Event
0	Reactor is operating at 105% NBR steam flow when LOP transient occurs initiating plant shutdown.
0	Concurrently loss of Division power (i.e., loss of one diesel generator) occurs.
15 ~~10~~ min.	Suppression pool cooling initiated to prevent overheating from SRV actuation.*
10 min.	Controlled blowdown initiated (100°F/hr) USING SELECTED SAFETY RELIEF VALVES
~~2-3 hrs.~~ 121 MIN.	Blowdown to 100 psi completed.
~~2-3 hrs.~~ 121 MIN.	Personnel are sent in to open RHR shutdown cooling suction valve; this fails.
~~2½-3½ hrs.~~ 151 MIN.	Actuate ADS and complete blowdown to suppression pool.
~~2½-3½ hrs.~~ 151 MIN.	Redirect RHR pump discharge from pool to vessel via LPCI line. Alternate cooling path now established.

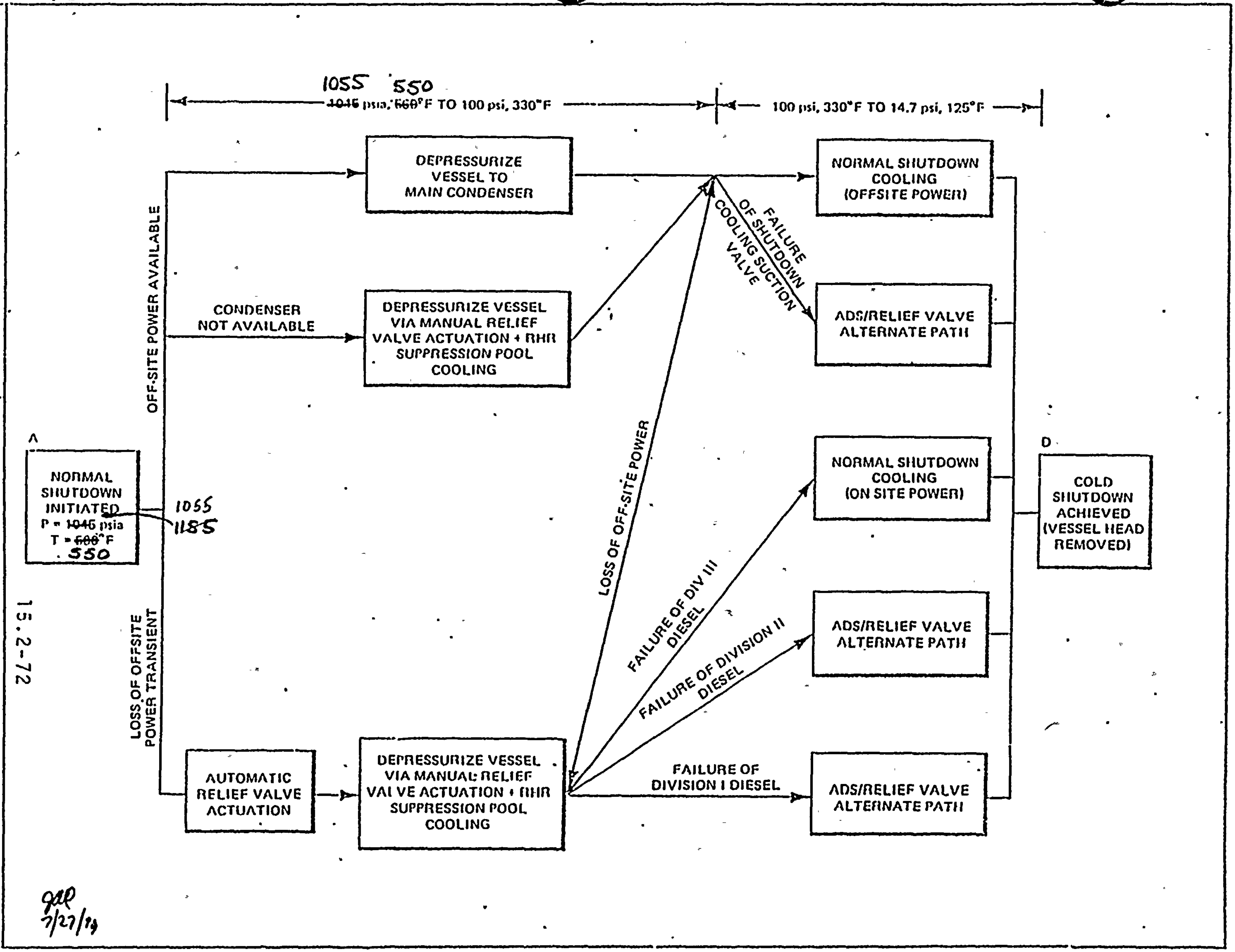
*See 15.2.6 for detailed sequence of events for loss of AC power transient.

7/21/75

Table ~~15.2.9-2~~

INPUT PARAMETERS FOR EVALUATION OF FAILURE OF RHR SHUTDOWN COOLING

Initial Power Corresponding To	105% Rated Steam Flow
Suppression Pool Mass (lbm)	8.52 E6
RHR (KHX value) (Btu/sec/°F)	289
Initial vessel condition	
Pressure (psia)	1055
Temperature (°F)	550.7
Initial primary fluid inventory (lbm)	7.016×10^5
Initial pool temperature (°F)	95
Service water temperature (°F)	87
Vessel heat capacity (Btu/lbm/°F)	0.123
HPCS on-off water level (ft)	
HPCS ON	40.8
HPCS OFF	47
HPCS flow rate (lbm/sec)	868
LPCI flow rate (lbm/sec)	982

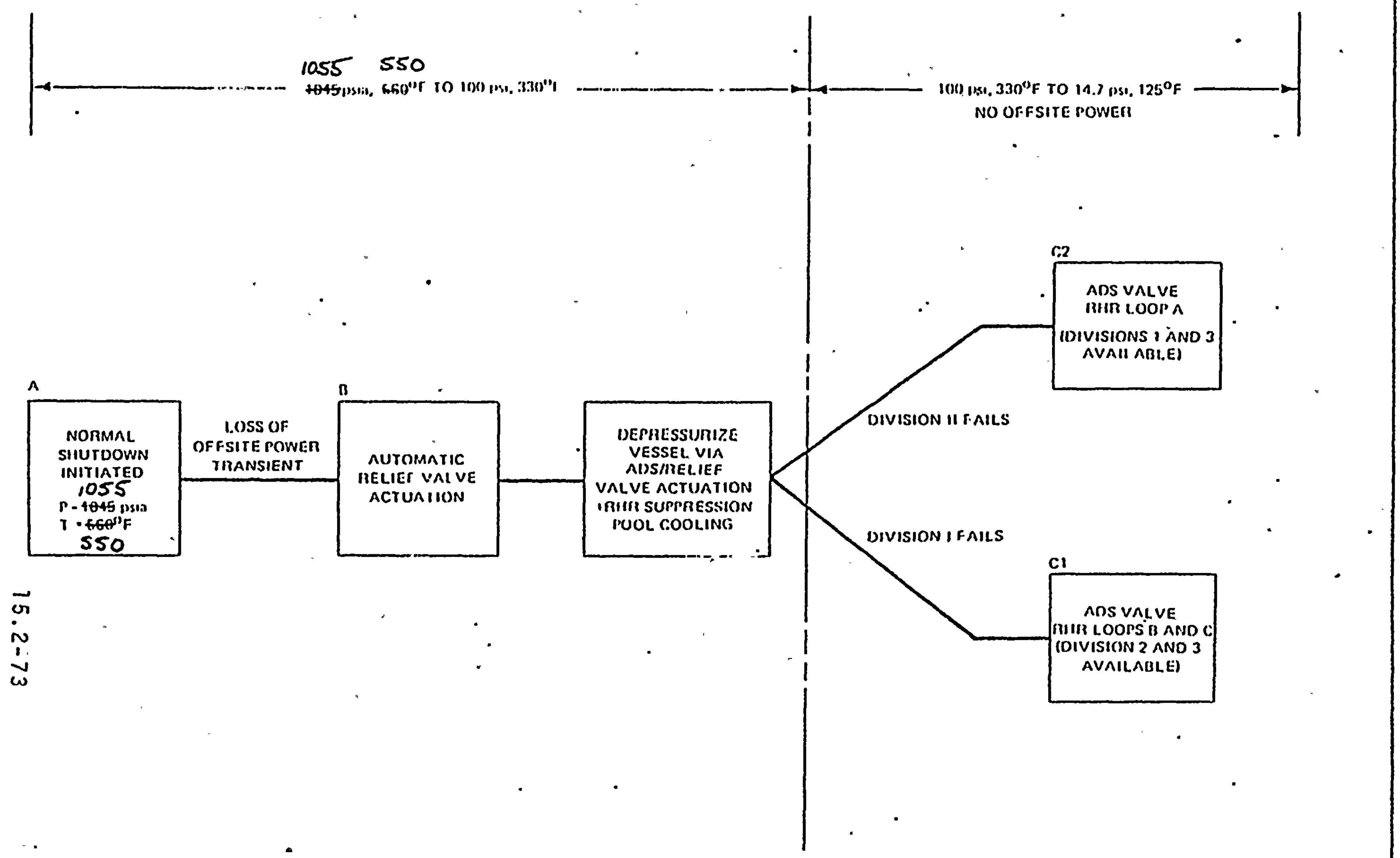


15.2-72

... PUBLIC POWER SUPPLY SYSTEM

SUMMARY OF PATHS AVAILABLE TO ACHIEVE ...

FIGURE



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WASHINGTON PUBLIC POWER SUPPLY SYSTEM
NUCLEAR PROJECT NO. 2

ADS/RHR COOLING LOOPS.

FIGURE 15.2-73

NOTES FOR FIGURE 15.2-11

ACTIVITY A

Initial Pressure = ~~1045~~ 1055 psia
Initial temperature = ~~560~~ 550°F

For purposes of this analysis, the following worst-case conditions are assumed to exist:

a. The reactor is assumed to be operating at 105% nuclear boiler rated steam flow;

b. A loss of power transient occurs (see 15.2.6), and

c. A simultaneous loss of onsite power (Division 1 or Division 2).

d. Operator unable to open one of the RHR shutdown cooling line suction valves.

ACTIVITY B

Initial system pressure = ~~1045~~ 1055 psia
Initial system temperature = ~~560~~ 550°F

Operator Actions

During approximately the first 30 minutes, reactor decay heat is passed to the suppression pool by the automatic operation of the reactor relief valves. Reactor water level will be returned to normal by the HPCS and RCIC system automatic operation.

After approximately 10 minutes, ~~it is assumed one RHR heat exchanger will be placed in the suppression pool cooling mode to remove decay heat.~~ The operator ~~will then~~ initiates depressurization of the reactor vessel to control vessel pressure. Controlled depressurization procedure consists of controlling vessel pressure and water level by using the SRV, RCIC and/or HPCS systems. AFTER APPROXIMATELY 15 MINUTES, IT IS ASSUMED ONE RHR HEAT EXCHANGER IS PLACED IN THE SUPPRESSION POOL COOLING MODE TO REMOVE DECAY HEAT. AT THIS TIME, THE SUPPRESSION POOL WILL BE 121°F. When the reactor pressure approaches 100 psig, the operator would normally prepare for operation of the RHR system in the shutdown cooling mode. AT THIS TIME (121 MIN), THE SUPPRESSION POOL WILL BE 186°F.

ACTIVITY C1 (Division 1 fails, Division 2 available)

System pressure = 100 psig
System temperature = 330°F

Operator Actions

The operator establishes a closed cooling path as follows:

a. Three to five ADS valves (DC Division 2) are powered open.

b. Either of the following cooling paths are established:

1. Utilizing RHR loop B, water from the suppression pool is pumped through the RHR heat exchanger (where a portion of the decay heat is removed) into the reactor vessel. The cooled suppression pool water flows through the vessel (picking up a portion of the decay heat) out the ADS valves and back to the suppression pool. This alternate cooling path is shown in Figure 15.2-12.

2. Utilizing RHR loops B and C together, water is taken from the suppression pool and pumped directly into the reactor vessel. The water passes through the vessel (picking up decay heat) and out the ADS valves returning to the suppression pool as shown in Figure 15.2-13. Suppression pool water is then cooled by operation of RHR loop B in the pool cooling mode (see Figure 15.2-14.). In this alternate cooling path RHR loop C is used for injection and RHR loop B for cooling. COLD SHUTDOWN IS ACHIEVED APPROXIMATELY 36 HOURS AFTER THE TRANSIENT OCCURS.

ACTIVITY C2 (Division 2 fails, Division 1 available) (Figure 15.2-15)

System pressure = 100 psig
System temperature = 330°F

Operator Actions

The operator establishes a closed cooling path as follows:

a. Three to five ADS valves (DC Division 1) are powered open.

b. Utilizing RHR loop A instead of loop B, an alternate cooling path is established as in Activity C1 item b.1 above. COLD SHUTDOWN IS REACHED IN APPROXIMATELY 15 HOURS.

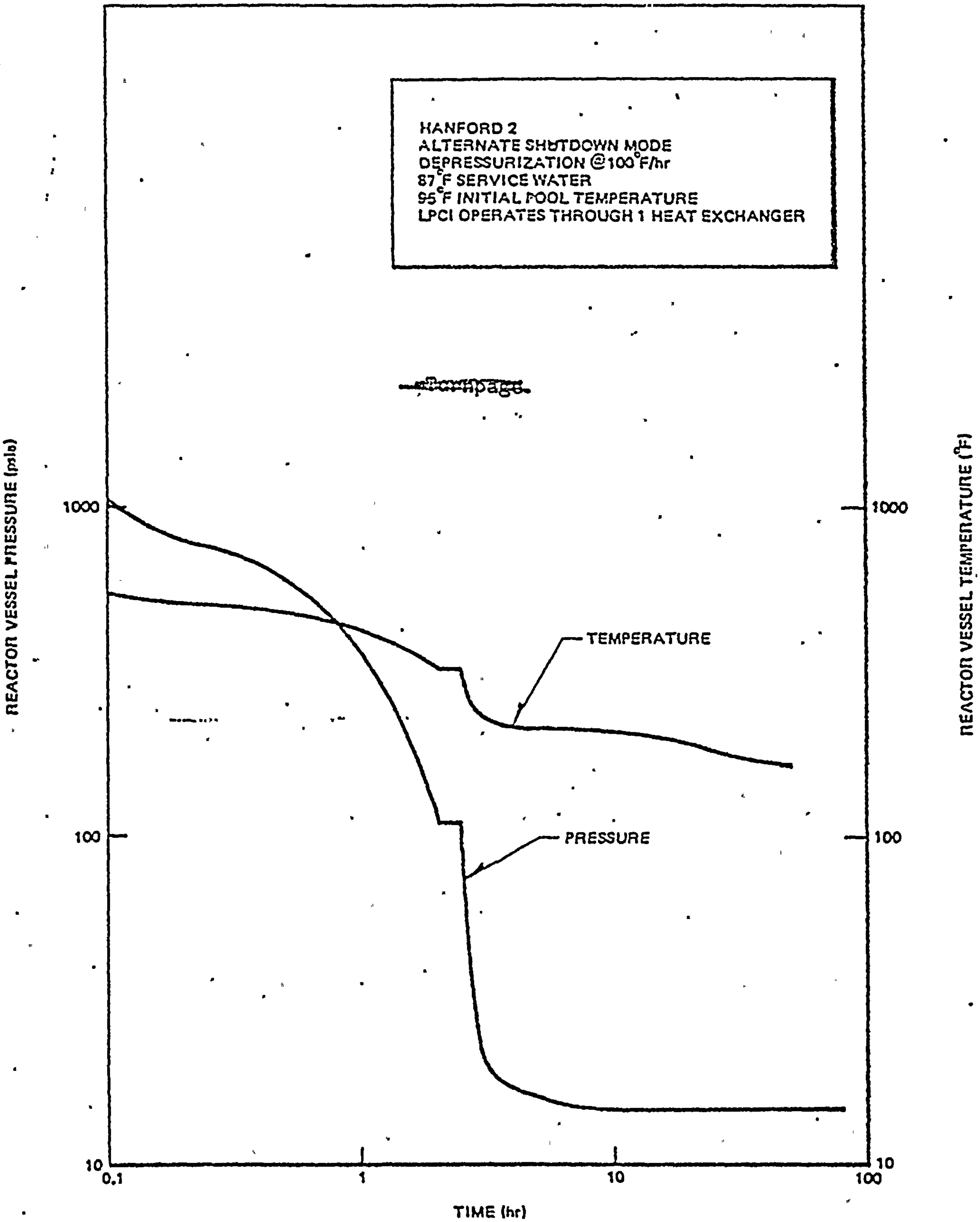


Figure ~~15.2.9-2b~~ 15.2-16

Vessel Temperature and Pressure Versus Time

(Activity C1~~(?)~~ .b.1 OR C2)

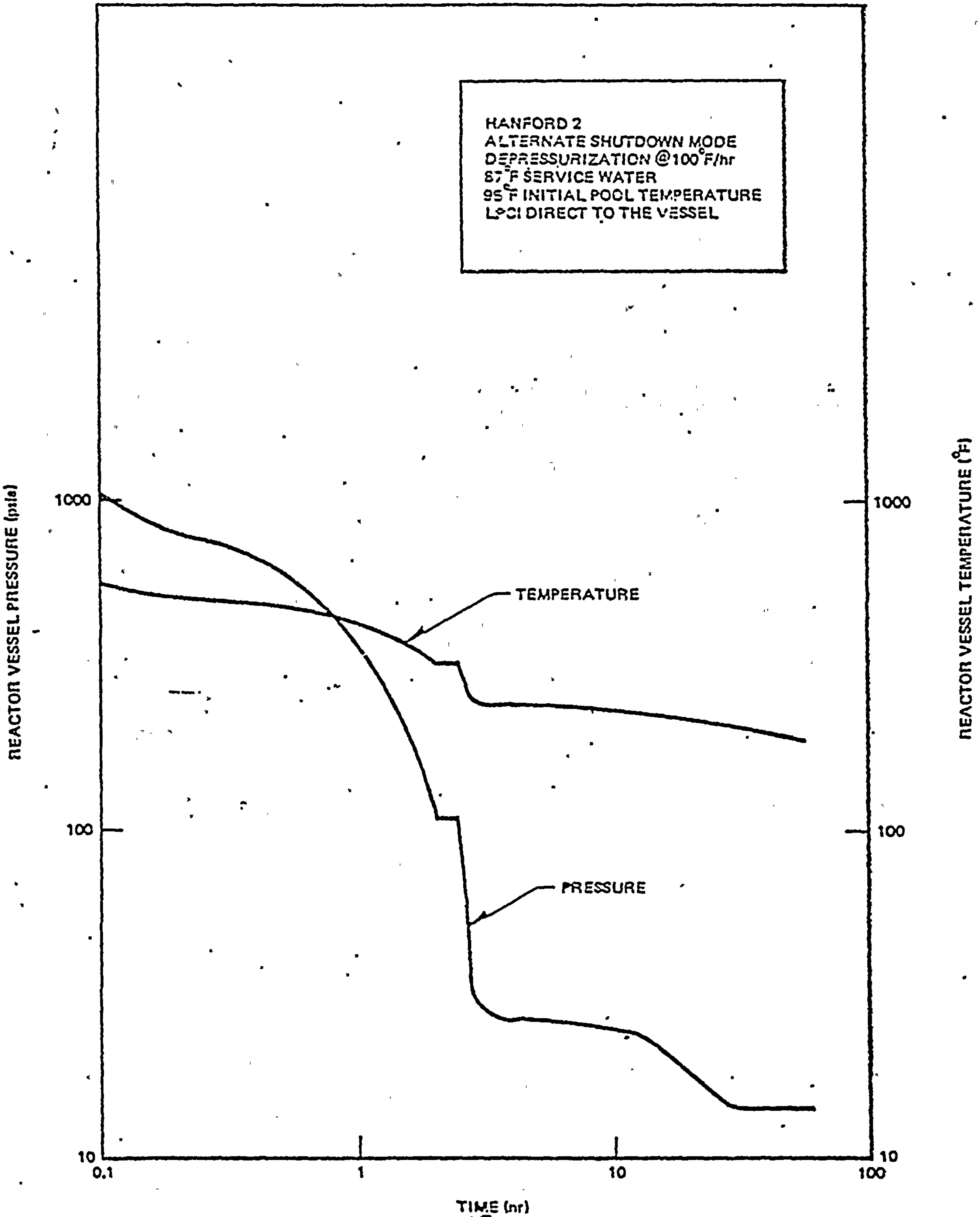


Figure 15-17 ~~15.2.9-2a~~

Vessel Temperature and Pressure Versus Time

(Activity C1.6.2 ~~(b)~~)

15.2-~~99a~~ 806

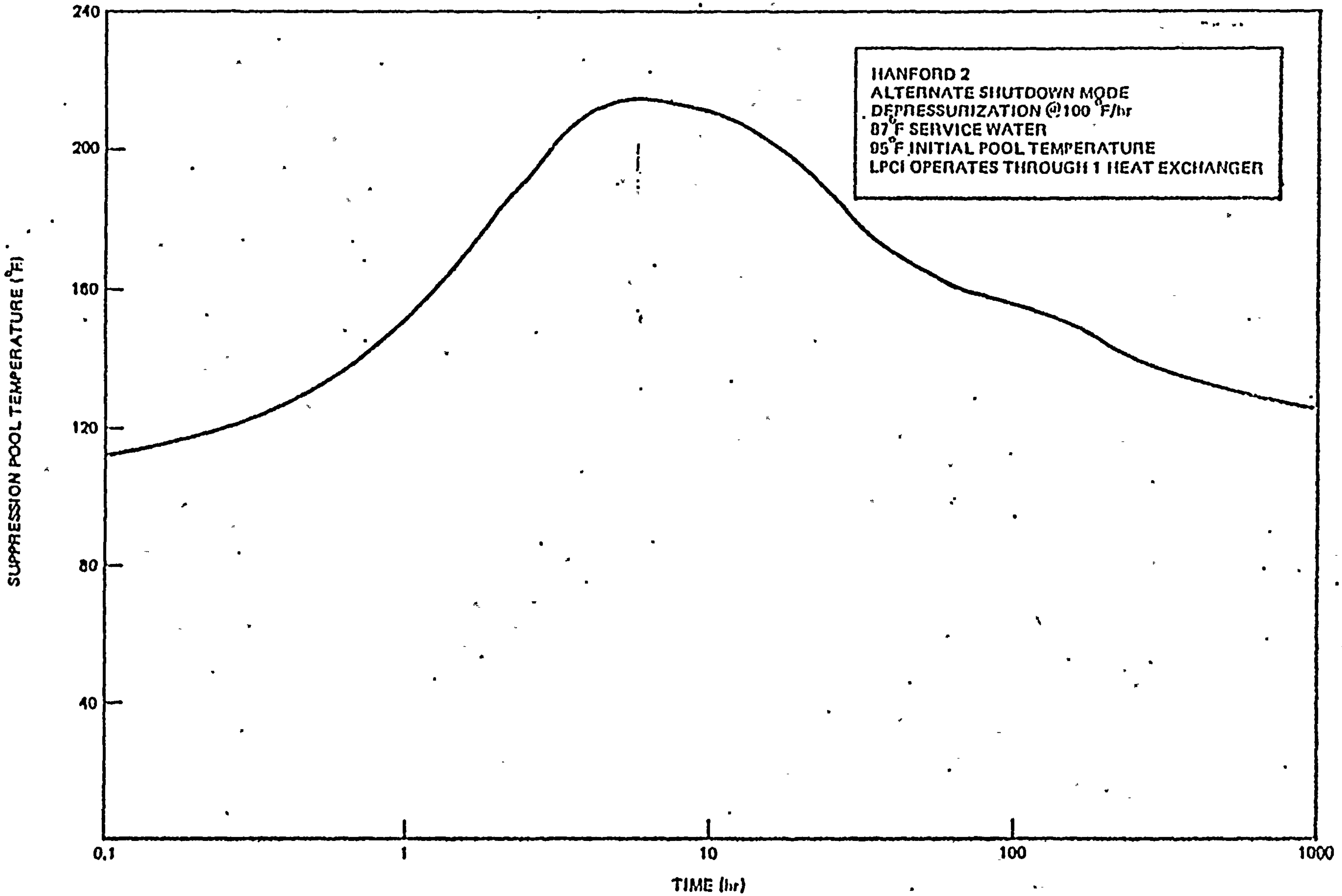


Figure ~~15.2.9-3a~~ 15.2-18

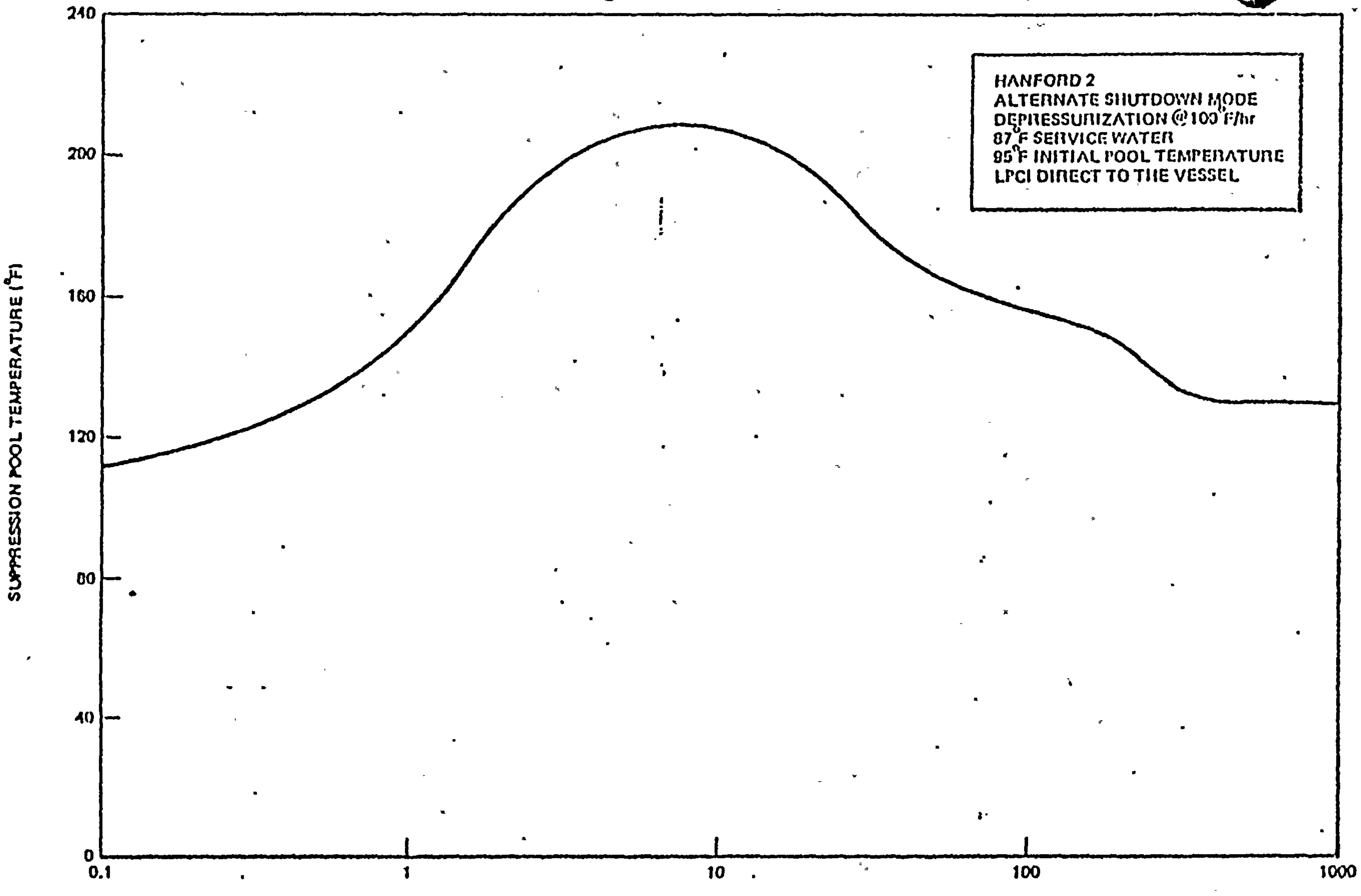
Suppression Pool Temperature Versus Time

(with 87°F Service Water Temperature)

(Activity C1(a) or C2)

.b.1

15.2-~~90a~~ 80c



15.2-19
Figure ~~15.2.9-3b~~

Suppression Pool Temperature Versus Time

(with 87°F Service Water Temperature)

(Activity C1 .b.2)

TABLE OF CONTENTS (Continued)

TABLES

TABLE OF CONTENTS (Continued)

FIGURES

Q. 211.048
(5.2.2)
(5.4.7)
(6.3)

In the FSAR, you state that the volume of air stored in the pneumatic accumulator for each safety/relief valves. You also state that the accumulator volume is sufficient for two actuations of the automatic depressurization system (ADS) valves. However, a "noninterruptible" safety-grade source of air to actuate the ADS valves is required to terminate certain postulated transients and accidents without loss of the ADS function. Demonstrate that an adequate supply of air will exist to operate the ADS valves for the following postulated accident conditions:

a. The alternate method of achieving and maintaining a cold shutdown following a loss of offsite power, concurrent with the worst single failure in the RHR system.

b. A small break LOCA concurrent with the failure of the high pressure core spray which would then require the ADS valves to be actuated to: (1) depressurize the reactor vessel; and (2) maintain long-term cooling. In your response, also discuss your proposed procedures to replenish the coolant inventory for this particular postulated accident.

c. A small steam line break disabling the RCIC concurrent with a single failure of the HPCS that would require actuation of the ADS to depressurize the reactor vessel. In your response, discuss the supply of air required to actuate the ADS valves to provide long-term cooling of the reactor core. Additionally, for this specific postulated accident condition, indicate whether the reactor vessel inventory would be maintained above the shutoff head of the low pressure cooling system when the decay heat of the reactor core repressurizes the vessel.

Response:

The ADS valves are capable of remaining open continuously during all postulated post-accident periods. For a description of the "noninterruptible" safety-grade source of air to the ADS valves, please refer to Section 9.3.1 and Figure 9.3-2. In an effort to clarify the FSAR text, Sections 5.2.2.4 and 9.3.1 have been modified.*

WNP-2

With regard to your request for additional information in Item 2 of this question, please refer to FSAR sections 6.3.2.2.3 and 6.3.2.2.4. Additionally, in response to Item 3, the vessel will not repressurize as you stated, since once the ADS valves open, they remain open.

*Draft FSAR page changes attached.

When the piston is actuated, the delay time, maximum elapsed time between receiving the overpressure signal at the valve actuator and the actual start of valve motion, will not exceed 0.1 seconds. The maximum full stroke opening time will not exceed 0.15 seconds.

The safety/relief valves can be operated in the power actuated mode by remote-manual controls from the main control room.

Each safety/relief valve is provided with its own pneumatic accumulator and inlet check valve. The accumulator capacity is sufficient to provide one safety/relief valve actuation, which is all that is required for overpressure protection. Subsequent actuations for an overpressure event can be spring actuations to limit reactor pressure to acceptable levels.

The safety/relief valves are designed to operate to the extent required for overpressure protection in the following accident environments:

a. 340°F for 3 hours at drywell design pressure

b. 320°F for an additional 3 hour period, at drywell design pressure

c. 250°F for an additional 18 hour period, at 25 psig

d. 200°F during the next 99 days at 20 psig. The duration of operability is two days following which the valves will remain fully open or closed for the remaining time period.

The Automatic Depressurization System (ADS) utilizes selected safety/relief valves for depressurization of the reactor, see 6.3, "Emergency Core Cooling System." Each of the safety/relief valves utilized for automatic depressurization is equipped with an air accumulator and check valve arrangement. These accumulators assure that the valves can be held open following failure of the air supply to the accumulators. They are sized to be capable of opening the valves and holding them open against the maximum drywell pressure of 45 psig. ~~The accumulator capacity is sufficient to provide two actuations for each ADS valve against 70% of maximum drywell pressure.~~ For a discussion of the noninterruptible air supply to the ADS valves see section 9.3.1.

The air compressors, dryer filters, and supply piping for the containment instrument air system provide air for charging the accumulators of the main steam isolation valves (inside primary containment) and the main steam safety/relief valves during normal operation (see 5.2 and 5.4). Since operation of this portion of the equipment in the containment instrument air system is not required for safe shutdown of the reactor (see 7.3 and 7.4 for effects of loss of instrument air on the main steam isolation and safety/relief valves), the pressure containing components are designed and constructed in accordance with ASME Section VIII, and system piping is designed and fabricated in accordance with ANSI B31.1, Seismic Category II. (System piping supports are Seismic Category I.) The only exception is that portion of the piping system from the outermost containment isolation valve to the solenoid valves of the main steam safety/relief and isolation valves (inside primary containment) which are ASME Section III Class 2, Safety Class 2 and Seismic Category I.

Two banks of nitrogen gas bottles and supply piping are provided as part of the containment instrument air system to supply the Automatic Depressurization System (ADS) main steam safety/relief valves with a pneumatic supply. ~~sufficient for 15 days following a postulated loss-of-coolant accident.~~ For a discussion of the function and operation of the ADS, see 6.3 and 7.3. The nitrogen bottles and associated equipment are classified Safety Class G, and the supply piping is ASME Section III, Class 2 and Class 3.

9.3.1.2 System Description

9.3.1.2.1 Control and Service Air System

The control and service air systems are shown schematically in Figure 9.3-1. Three compressors and three air receivers supply both control and service air requirements.

Each compressor has a start-standby-stop remote selector switch and an unloader control. When the selector switch is set in the start position, the compressor runs continuously and loads and unloads to maintain receiver pressure. When the selector switch is in standby position, the standby compressor will automatically start when receiver pressure falls below 90 psig. Normally two compressors are running and one compressor is placed on standby.

This system consists of two 100% capacity air compressors, associated coolers, a twin tower air dryer, filters, an air receiver, valves, and piping of a leak tight design. In addition, two nitrogen gas bottle banks and associated piping are provided as a backup to the compressor supplied air for seven of the main steam relief valves which perform the ADS function.

The compressors located in the reactor building take suction from the building atmosphere through intake filter-silencers which are 98% efficient in filtration of particles as fine as five microns. The air is then discharged through an aftercooler, a prefilter, a dryer, an afterfilter and air receiver to deliver dry, clean, pressurized air to the pneumatic control systems of the following valves inside the primary containment vessel:

a. Four main steam isolation valves and their accumulators,

b. Eighteen main steam safety/relief valves and their accumulators.

The two independent nitrogen bottle bank subsystems are provided to deliver pressurized nitrogen to seven of the safety/relief valves and accumulators. ~~located in the nitrogen piping inside the primary containment.~~ These seven valves perform the ADS function, if required, during postulated LOCA ~~accident~~ conditions. These nitrogen banks ensure a 30 day supply of nitrogen for the ADS function during isolation of the compressor loop. One ~~bottle~~ bank of 15 bottles ~~system~~ supplies nitrogen for three main steam safety/relief valves and accumulators, while the other bank of 19 bottles supplies four main steam safety/relief valves and accumulators (see Figure 9.3-2).

[Add Insert "A" here]

9.3.1.3 Safety Evaluation

9.3.1.3.1 Control and Service Air System

Operation of the control and service air system is not required for the initiation of any engineered safeguard systems or for safe shutdown of the reactor.

For FSAR p. 9.3-4

Insert "A"

These nitrogen bottles are located in the railroad lock of the reactor building to facilitate access. The bottles are standard, commercially available units pressurized to 2490 psig. Each bottle has a capacity of 257SCF. The required quantity of bottles for each bank was conservatively based on providing a 30 day supply to the ADS valves to satisfy the long term post-LOCA demand based on the following:

N_2 bank	N_2 Supply	System Leakage *	ADS cyles allowed **
A (3 valves)	3855 SCF	2880 SCF	48
B (4 valves)	4883 SCF	3600 SCF	48

* Based on 1SCFH leakage per ADS valve and an additional 1SCFH leakage for each valve group

** Based on a requirement of 6.7 SCF per valve to open the valve with zero reactor pressure and maximum drywell pressure.

Once opened, the ADS valves are not expected to be cycled during the post-accident period; nevertheless, the air supply was conservatively sized to allow for numerous cycles.

9.3.1.3.2 Containment Instrument Air System

Since each of the two nitrogen supplies and the compressed air supply are independent of each other, a single component failure in one will not effect the operational function of the other.

During normal operation, one compressor will operate intermittently to restore loss of pressure in the main steam isolation and safety/relief valve accumulators. The compressor loop discharge piping can be isolated, if required, under accident conditions. Each nitrogen bottle supply line isolation valve is powered from a different division of the critical power supply.

In the event of loss of power to the compressed air supply, the individual air accumulators serve as a reliable source of compressed air for the main steam isolation and safety/relief valves. ~~The nitrogen supply systems provide a 15-day supply of nitrogen to seven main steam safety/relief valves and accumulators for the ADS function during postulated accident conditions and/or power loss, if required.~~ Further discussion of the effects of loss of air to the main steam isolation and safety/relief valves is presented in 7.3 and 7.4.

9.3.1.4 Testing and Inspection Requirements

The systems are inspected and cleaned prior to service. Instruments are calibrated during testing, and automatic controls are tested for actuation at the proper set points. Alarm functions are checked for operability and limits during plant operational testing. The systems are operated and tested initially with regard to flow paths, flow capacity, and mechanical operability in accordance with Chapter 14.

The air compressors normally in operation will be selected based upon a rotating schedule to equalize operating time. The rotation of operation also acts as an operational test of the compressor. Conformance to Regulatory Guide 1.5. (ASME Code, Section XI) is discussed in 6.6.

9.3.1.5 Instrumentation Requirements

RESPONSES TO

ANALYSIS BRANCH (AB II)

QUESTIONS 222.001 - 222.004

Q. 222.001
(6.2.1)

Describe in detail how you evaluated the mass and energy release data during the complete blowdown phase (i.e., during the first 100 seconds) for a postulated break in the recirculation line and in the feedwater line. Describe all analytical models which you used, including your assumptions. If any hand calculations were performed, provide the detailed calculations.

Response:

A detailed description of the analytical models and assumptions for a recirculation line break and subsequent blowdown is described in References 6.2-1 and 6.2-2 of Section 6.2.7.

The feedwater line break and subsequent blowdown is associated with annulus pressurization and is addressed in the response to Question 222.002.

Q. 222.002

Provide a detailed description of your analytical model to evaluate the mass and energy release rates for your analyses of the short-term annulus pressurization and the evaluation of the structural loads resulting from postulated pipe breaks for the first five seconds following the accident. Indicate the mass flux ($LB_M/sec-ft^2$), the enthalpy (BTU/LB_M) and the flow area (square feet) as a function of time for each side of the break. Justify all your assumptions. Describe the break geometry assumed throughout the transient. Discuss the overall conservatism of your analysis.

Response:

Extensive documentation has been submitted by WPPSS to NRC concerning mass and energy release rates for short-term annulus pressurization in response to a post-construction permit item on the sacrificial shield design. Please refer to references 3.8-5, 3.8-6, and 3.8-7 of the FSAR for the requested information (referenced from Sections 3.8.3.1.2 and 6.2.1.2). Copies of these references have been submitted to the NRC before and more recently to Mr. Jack Kudrick of Containment Systems Branch via Reference 1. The NRC in References 2 and 3 found the WPPSS reports acceptable. The information may be updated in the future, however, to reflect changes necessitated by current piping system analysis. As such, these revisions will be reflected in amendments to Section 6.2.1.2 of the FSAR.

In summary, though, for the short-term annulus pressurization analysis and subsequent evaluation of structural loads the analytical model to evaluate mass and energy release rates is the simple and conservative Moody's two phase critical flows model. For a break in LPCI, HPCS, LPCS, feedwater, and recirculation lines, the following assumptions are used:

a. The break is the double-ended guillotine type which opens instantaneously.

b. For the time of interest (the first few seconds of the break) the blowdown flow rate and energy are constant.

c. Water from the reactor side of the break is saturated water at 1060 psia and enthalpy of 550 Btu/lb.

d. Moody's critical flows rate is 8100 $lb_m/sec/ft^2$ for the conditions listed in .

e. Reactor depressurization or change in flow quality is not considered.

f. Initial fluid inventory in the pipe is depleted at a rate consistent with the enthalpy of the fluid in the pipe.

g. Frictional loss of flow in the pipe is neglected.

The constant flow approach is used because it is more conservative than the time dependent blowdown calculation.. Assumptions a, e and g are also intended to maximize the break flow rate, and therefore to produce conservative results. For additional conservatism, the enthalpy of flow from the pipe side is taken to be the same as the enthalpy of flow from the reactor side (550 Btu/lb).

References:

1. Letter, D. L. Renberger (WPPSS) to L. Rubenstein (NRC), "Provision of Sacrificial Shield Wall Documents and Drawings", G02-80-10, January 10, 1980.

2. Letter, R. C. DeYoung (NRC) to J. J. Stein (WPPSS), August 13, 1975.

3. Letter, R. C. DeYoung (NRC) to J. J. Stein (WPPSS), October 15, 1975.

Q. 222.003

Describe in detail how the long-term steaming rates were developed for the time period following a postulated loss-of-coolant accident (LOCA). If the steaming rates were developed by hand calculations, provide the details of your method and list your assumptions. Describe the break flow area as a function of time. Discuss the overall conservatism of your analysis.

Response:

Once the reactor pressure vessel has been reflooded to the break elevation during the short-term response, energy is removed from the vessel by water flowing out from the break at a rate equal to the ECCS flow rate into the vessel (Ref: Section 6.2.1.1.3.4.2). The enthalpy of the flow spilling from the break is determined from an energy balance, as described in Section 6.2.1.1.3.4.4. Since energy is removed from the vessel by way of a closed cooling loop of subcooled water flowing from the suppression pool to the vessel and returning to the pool through the break, long-term steaming would not occur.

This analysis is conservative for long-term containment cooling since removal of energy by steaming would require that more energy be retained in the vessel, and therefore not released to the containment, in order to maintain the vessel fluid inventory at saturation temperature.

The break flow area is assumed to remain constant as a function of time following decompression of the broken line and/or closure of the main steam isolation valves during the first few seconds of the reactor blowdown.