

Attachment 1

Technical Specification Changed Pages
in Support of the Pressure/Temperature Limits Report.

REMOVE

ii

iii

xx(a)

1-5

3/4 4-18

3/4 4-19

3/4 4-20

3/4 4-21

B 3/4 4-4

B 3/4 4-5

6-19

INSERT

ii

iii

xx(a)

1-5

1-5a

3/4 4-18

3/4 4-19

B 3/4 4-4

B 3/4 4-5

6-19



22. 0

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

100

CONTROLLED COPY

INDEX

DEFINITIONS

SECTION

DEFINITIONS (Continued)

PAGE

| | |
|--|------|
| 1.22 LOGIC SYSTEM FUNCTIONAL TEST..... | 1-4 |
| 1.23 MAXIMUM FRACTION OF LIMITING POWER DENSITY..... | 1-4 |
| 1.24 MAXIMUM TOTAL PEAKING FACTOR..... | 1-4 |
| 1.25 MEMBER(S) OF THE PUBLIC..... | 1-4 |
| 1.26 MINIMUM CRITICAL POWER RATIO..... | 1-4 |
| 1.27 OFFSITE DOSE CALCULATION MANUAL..... | 1-4a |
| 1.28 OPERABLE - OPERABILITY..... | 1-5 |
| 1.29 OPERATIONAL CONDITION - CONDITION..... | 1-5 |
| 1.30 PHYSICS TESTS..... | 1-5 |
| 1.31 PRESSURE BOUNDARY LEAKAGE..... | 1-5 |
| 1.31 A PRESSURE/TEMPERATURE LIMITS REPORT | 1-5 |
| 1.32 PRIMARY CONTAINMENT INTEGRITY..... | 1-5a |
| 1.33 PROCESS CONTROL PROGRAM..... | 1-6 |
| 1.34 PURGE - PURGING..... | 1-6 |
| 1.35 RATED THERMAL POWER..... | 1-6 |
| 1.36 REACTOR PROTECTION SYSTEM RESPONSE TIME..... | 1-6 |
| 1.37 REPORTABLE EVENT..... | 1-6 |
| 1.38 ROD DENSITY..... | 1-6 |
| 1.39 SECONDARY CONTAINMENT INTEGRITY..... | 1-6 |
| 1.40 SHUTDOWN MARGIN..... | 1-7 |
| 1.41 SITE BOUNDARY..... | 1-7 |
| 1.42 NOT USED..... | 1-7 |
| 1.43 SOURCE CHECK..... | 1-7 |
| 1.44 STAGGERED TEST BASIS..... | 1-7 |

move
to
next
page

INDEX

DEFINITIONS

SECTION

DEFINITIONS (Continued)

PAGE

| | |
|--|-----|
| 1.45 THERMAL POWER..... | 1-8 |
| 1.46 TOTAL PEAKING FACTOR..... | 1-8 |
| 1.47 TURBINE BYPASS SYSTEM RESPONSE TIME..... | 1-8 |
| 1.48 UNIDENTIFIED LEAKAGE..... | 1-8 |
| 1.49 UNRESTRICTED AREA..... | 1-8 |
| 1.50 VENTILATION EXHAUST TREATMENT SYSTEM..... | 1-8 |
| 1.51 VENTING..... | 1-8 |

add
1.44
from
previous
page

101

102

103

104

105

106

107

108

109

110

111

112

113

114

115

116

117

118

119

120

121

122

123

124

125

INDEX

LIST OF FIGURES

| <u>FIGURE</u> | | <u>PAGE</u> |
|---------------|--|-----------------------------|
| 3.2.4-5 | LINEAR HEAT GENERATION RATE (LHGR) LIMIT VERSUS AVERAGE PLANAR EXPOSURE GELI LEAD FUEL ASSEMBLIES..... | Deleted |
| 3.2.6-1 | OPERATING REGION LIMITS OF SPEC. 3.2.6..... | 3/4 2-6 |
| 3.2.7-1 | OPERATING REGION LIMITS OF SPEC. 3.2.7..... | 3/4 2-8 |
| 3.2.8-1 | OPERATING REGION LIMITS OF SPEC. 3.2.8..... | 3/4 2-10 |
| 3.4.1.1-1 | THERMAL POWER LIMITS OF SPEC. 3.4.1.1-1..... | 3/4 4-3a |
| 3.4.6.1 | MINIMUM REACTOR VESSEL METAL TEMPERATURE VERSUS REACTOR VESSEL PRESSURE..... | 3/4 4-20 Deleted |
| 4.7-1 | SAMPLE PLAN 2) FOR SNUBBER FUNCTIONAL TEST | 3/4 7-15 |
| 3.9.7-1 | HEIGHT ABOVE SFP WATER LEVEL VS. MAXIMUM LOAD TO BE CARRIED OVER SFP..... | 3/4 9-10 |
| B 3/4 3-1 | REACTOR VESSEL WATER LEVEL..... | B 3/4 3-8 |
| B 3/4.4.6-1 | FAST NEUTRON FLUENCE ($E > 1\text{MeV}$) AT $1/4$ T AS A FUNCTION OF SERVICE LIFE..... | B 3/4 4-7 |
| 5.1-1 | EXCLUSION AREA BOUNDARY | 5-2 |
| 5.1-2 | LOW POPULATION ZONE..... | 5-3 |
| 5.1-3 | UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS..... | 5-4 |

OPERABLE - OPERABILITY

- 1.28 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL CONDITION - CONDITION

- 1.29 An OPERATIONAL CONDITION, i.e., CONDITION, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1.2.

PHYSICS TESTS

- 1.30 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation as (1) described in Chapter 14 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

- 1.31 PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall, or vessel wall.

PRIMARY CONTAINMENT INTEGRITY

- 1.32 PRIMARY CONTAINMENT INTEGRITY shall exist when:

- a. All primary containment penetrations required to be closed during accident conditions are either:
 1. Capable of being closed by an OPERABLE primary containment automatic isolation system, or
 2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position. except as provided in Table 3.6.3-1 of Specification 3.6.3.
- b. All primary containment equipment hatches are closed and sealed.
- c. Each primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.
- d. The primary containment leakage rates are within the limits of Specification 3.6.1.2.
- e. The suppression chamber is in compliance with the requirements of Specification 3.6.2.1.
- f. The sealing mechanism associated with each primary containment penetration; e.g., welds, bellows, or O-rings, is OPERABLE.

Insert to page 1-5

PRESSURE/TEMPERATURE LIMITS REPORT

- 1.31A The PRESSURE/TEMPERATURE LIMITS REPORT (PTLR) is the WNP-2 specific document that provides the reactor vessel pressure and temperature limits including heatup and cooldown rates for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 6.9.12. Plant operation with these operating limits is addressed in LCO 3.4.6.1 (REACTOR COOLANT SYSTEM PRESSURE/TEMPERATURE LIMITS).



1.1

2

3

4

5

6

7

8

9

10

11

12

13

14

15

16

17

18

19

20

21

22

23

CONTROLLED COPY

REACTOR COOLANT SYSTEM

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

provided in the PRESSURE/TEMPERATURE LIMITS REPORT during heatup, cooldown, criticality, and inservice leak and hydrostatic testing.

3.4.6.1 The reactor coolant system temperature and pressure shall be limited in accordance with the limits lines shown on Figure 3.4.6.1 (1) curve A for hydrostatic or leak testing; (2) curve B for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and (3) curve C for operations with a critical core other than low power PHYSICS TESTS, with:

- a. A maximum heatup of 100°F in any 1-hour period,
- b. A maximum cooldown of 100°F in any 1-hour period,
- c. A maximum temperature change of less than or equal to 20°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and
- d. The reactor vessel flange and head flange temperature greater than or equal to 80°F when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

limits as specified in the PRESSURE/TEMPERATURE LIMITS REPORT

4.4.6.1.1 During system heatup, cooldown, and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and to the right of the limit lines of Figure 3.4.6.1 curves A, B or C, as applicable, at least once per 30 minutes.

1. The first part of the document is a list of names and addresses of the members of the committee.

2. The second part of the document is a list of names and addresses of the members of the committee.

3. The third part of the document is a list of names and addresses of the members of the committee.

4. The fourth part of the document is a list of names and addresses of the members of the committee.

5. The fifth part of the document is a list of names and addresses of the members of the committee.

6. The sixth part of the document is a list of names and addresses of the members of the committee.

7. The seventh part of the document is a list of names and addresses of the members of the committee.

CONTROLLED COPY

REACTOR COOLANT SYSTEM

WNP-2 Pressure/Temperature Limits curve C of the
PRESSURE/TEMPERATURE LIMITS REPORT

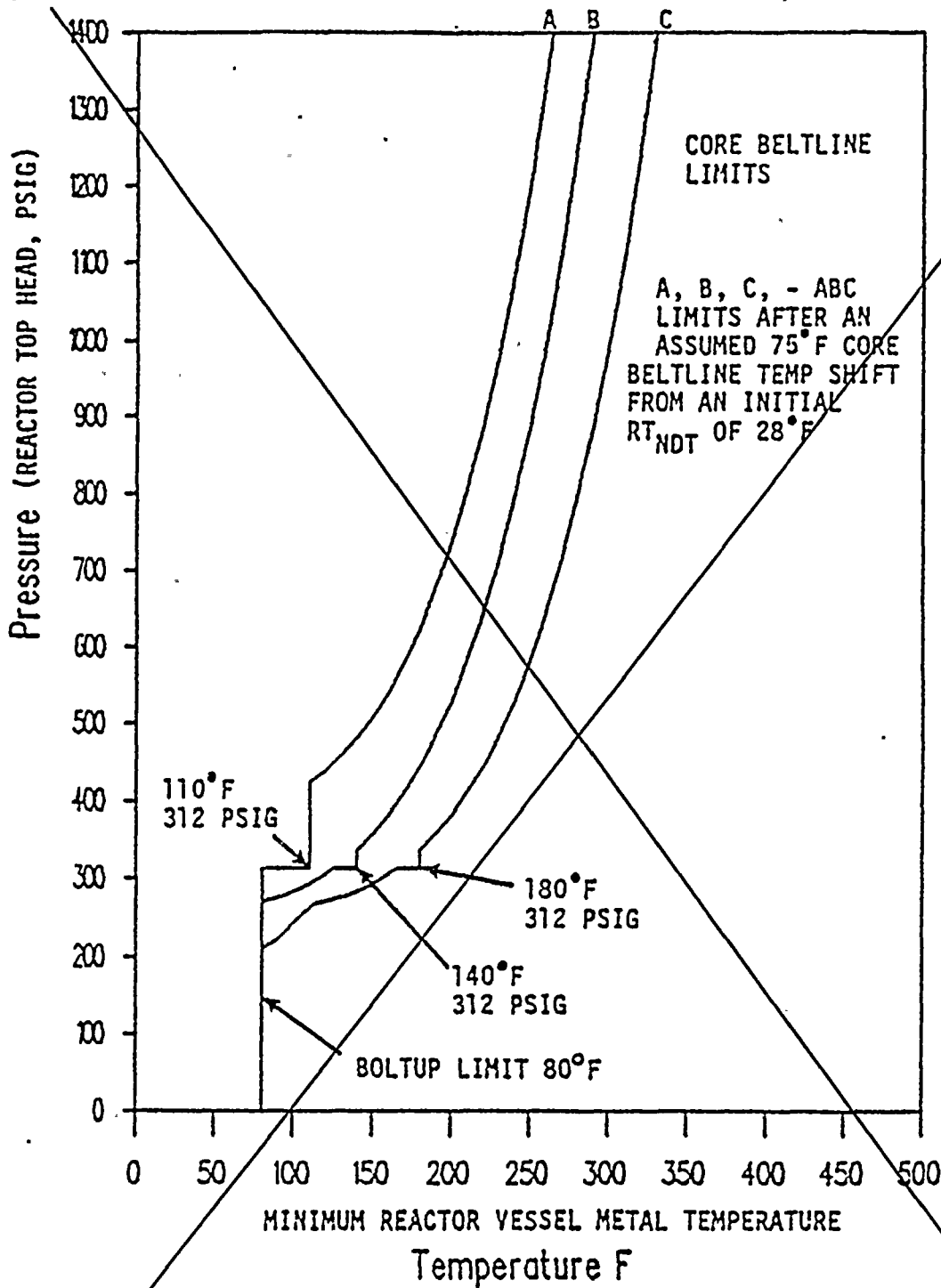
SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the ~~criticality limit line of Figure 3.4.6.1 curve~~ within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and at least once per 30 minutes during system heatup.

4.4.6.1.3 The reactor vessel material surveillance specimens shall be removed and examined, to determine changes in reactor pressure vessel material properties as required by 10 CFR Part 50, Appendix H in accordance with the schedule in ~~Table 4.4.6.1.3-1~~. The results of these examinations shall be used to update the curves of Figure ~~3.4.6.1-1~~ of the PRESSURE/TEMPERATURE LIMITS REPORT. NRC approved

4.4.6.1.4 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 80°F:

- a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:
1. $\leq 100^{\circ}\text{F}$, at least once per 12 hours.
 2. $\leq 90^{\circ}\text{F}$, at least once per 30 minutes.
- b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.



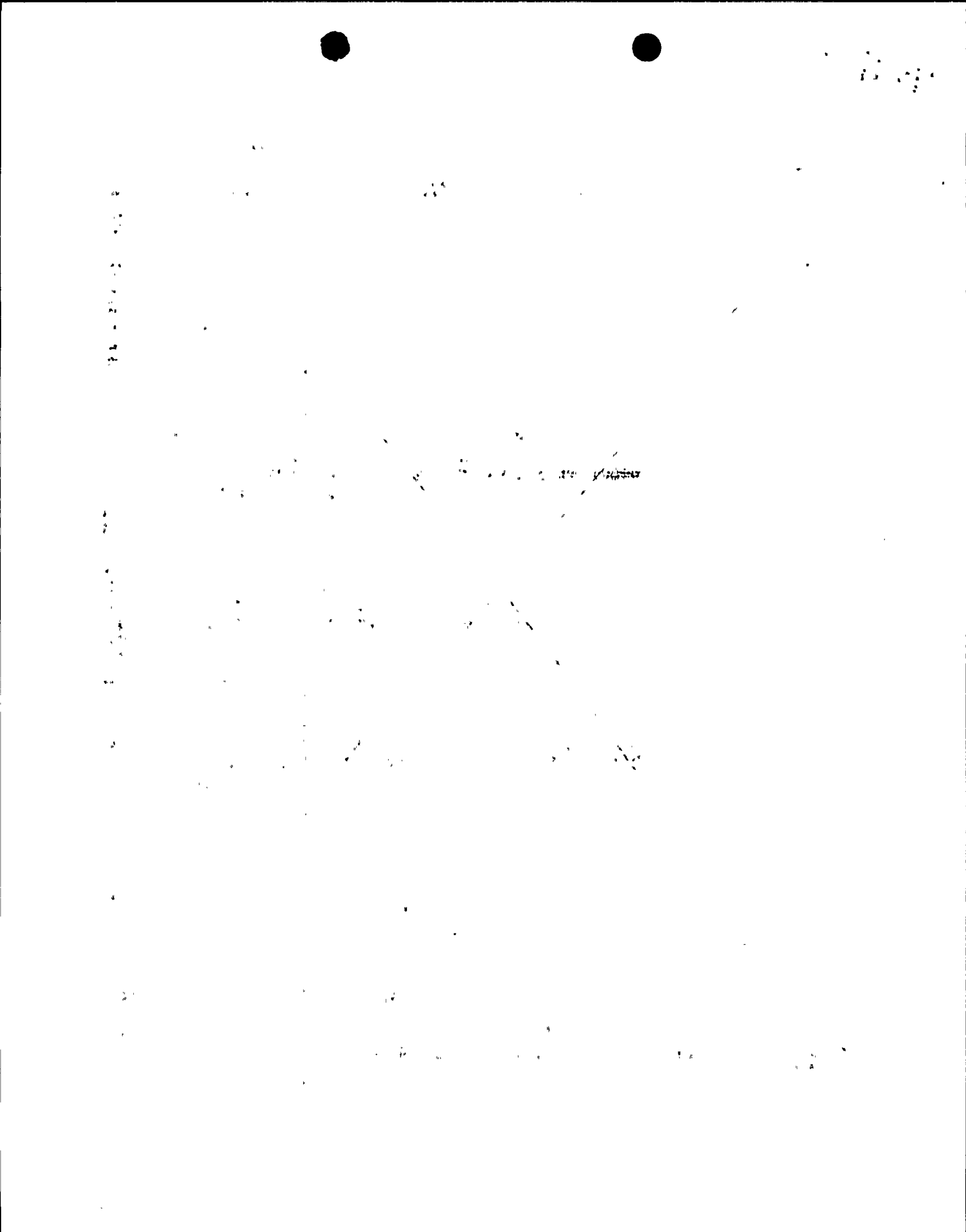
Relocated to
PRESSURE/TEMPERATURE
LIMITS REPORT

FIGURE 3.4.6.1

MINIMUM REACTOR VESSEL METAL TEMPERATURE VERSUS
REACTOR VESSEL PRESSURE



THIS PAGE IS BLANK INTENTIONALLY



REACTOR COOLANT SYSTEM

BASES

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.9 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady-state conditions, i.e., no thermal stresses, represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

The reactor vessel materials have been tested to determine their initial RT_{NDT} . The results of these tests are shown in Table B 3/4.4.6-1. Reactor operation and resultant fast neutron irradiation, E greater than 1 MeV, will cause an increase in the RT_{NDT} . Therefore, an adjusted reference temperature, based upon the fluence, nickel content, and copper content of the material in question, can be predicted using Bases Figure B 3/4.4.6-1 and the recommendations of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." The pressure/temperature limit curve, ~~Figure 3.4.6-1~~ includes predicted adjustments for this shift in RT_{NDT} . ~~for the end of life fluence and is effective for 10 EFPY.~~
 provided in the PRESSURE/TEMPERATURE LIMITS REPORT

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and 10 CFR Part 50, Appendix H, irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel in the core area. The irradiated specimens can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of ~~Figure 3.4.6-1~~ shall be adjusted, as required, on the basis of the specimen data and recommendations of Regulatory Guide 1.99, Revision 2.

the PRESSURE/TEMPERATURE LIMITS REPORT

10-11-68

10-11-68

10-11-68

10-11-68

10-11-68

10-11-68

10-11-68

CONTROLLED COPY

REACTOR COOLANT SYSTEM

BASES

PRESSURE/TEMPERATURE LIMITS (Continued)

the PRESSURE/TEMPERATURE LIMITS
REPORT

The pressure-temperature limit lines shown in ~~Figure 3.4.6-1~~ for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50 for reactor criticality and for inservice leak and hydrostatic testing.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment, however, single failure considerations require that two valves be OPERABLE. The surveillance requirements are based on the operating history of this type valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks. The minimum closure time is consistent with the assumptions in the safety analyses to prevent pressure surges.

3/4.4.8 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant.

Access to permit inservice inspections of components of the reactor coolant system is in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1974 Edition and Addenda through Summer 1975.

The inservice inspection program for ASME Code Class 1, 2 and 3 components will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the NRC pursuant to 10 CFR 50.55a(g)(6)(i).

3/4.4.9 RESIDUAL HEAT REMOVAL

A single shutdown cooling mode loop provides sufficient heat removal capability for removing core decay heat and mixing to assure accurate temperature indication, however, single failure considerations require that two loops be OPERABLE or that alternate methods capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation.

RADIOACTIVE EFFLUENT RELEASE REPORT

6.9.1.11 The Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a(a)(2).

The Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

Insert from following page.

12.12.77

12.12.77

12.12.77

12.12.77

12.12.77

12.12.77

Insert to page 6-19:

PRESSURE/TEMPERATURE LIMITS REPORT

- 6.9.1.12 The PRESSURE/TEMPERATURE LIMITS REPORT is the unit-specific document that provides the reactor vessel pressure and temperature limits including heatup and cooldown rates for the current reactor vessel fluence period. Plant operation within these operating limits is addressed in Specification 3.4.6.1 (Pressure/Temperature Limits).

Pressure and temperature limits shall be established and documented in the PRESSURE/TEMPERATURE LIMITS REPORT before each reactor vessel fluence period or any remaining part of a reactor vessel fluence period for Specification 3.4.6.1, Pressure/Temperature Limits. The analytical methods used to determine the pressure and temperature limits shall be those previously reviewed and approved by the NRC. The pressure/temperature limits shall be determined so that all applicable limits of the safety analysis are met. The PRESSURE/TEMPERATURE LIMITS REPORT, shall be provided upon issuance, for each reactor vessel fluence period to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

Changes to this report shall become effective after review and acceptance by the Plant Operating Committee and the approval of the Plant Manager.

24

1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
17
18
19
20
21
22
23
24
25
26
27
28
29
30
31
32
33
34
35
36
37
38
39
40
41
42
43
44
45
46
47
48
49
50
51
52
53
54
55
56
57
58
59
60
61
62
63
64
65
66
67
68
69
70
71
72
73
74
75
76
77
78
79
80
81
82
83
84
85
86
87
88
89
90
91
92
93
94
95
96
97
98
99
100

Attachment 2

Example of the Pressure/Temperature Limits Report

12-23-72

1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
17
18
19
20
21
22
23
24
25
26
27
28
29
30
31
32
33
34
35
36
37
38
39
40
41
42
43
44
45
46
47
48
49
50
51
52
53
54
55
56
57
58
59
60
61
62
63
64
65
66
67
68
69
70
71
72
73
74
75
76
77
78
79
80
81
82
83
84
85
86
87
88
89
90
91
92
93
94
95
96
97
98
99
100
101
102
103
104
105
106
107
108
109
110
111
112
113
114
115
116
117
118
119
120
121
122
123
124
125
126
127
128
129
130
131
132
133
134
135
136
137
138
139
140
141
142
143
144
145
146
147
148
149
150
151
152
153
154
155
156
157
158
159
160
161
162
163
164
165
166
167
168
169
170
171
172
173
174
175
176
177
178
179
180
181
182
183
184
185
186
187
188
189
190
191
192
193
194
195
196
197
198
199
200
201
202
203
204
205
206
207
208
209
210
211
212
213
214
215
216
217
218
219
220
221
222
223
224
225
226
227
228
229
230
231
232
233
234
235
236
237
238
239
240
241
242
243
244
245
246
247
248
249
250
251
252
253
254
255
256
257
258
259
260
261
262
263
264
265
266
267
268
269
270
271
272
273
274
275
276
277
278
279
280
281
282
283
284
285
286
287
288
289
290
291
292
293
294
295
296
297
298
299
300
301
302
303
304
305
306
307
308
309
310
311
312
313
314
315
316
317
318
319
320
321
322
323
324
325
326
327
328
329
330
331
332
333
334
335
336
337
338
339
340
341
342
343
344
345
346
347
348
349
350
351
352
353
354
355
356
357
358
359
360
361
362
363
364
365
366
367
368
369
370
371
372
373
374
375
376
377
378
379
380
381
382
383
384
385
386
387
388
389
390
391
392
393
394
395
396
397
398
399
400
401
402
403
404
405
406
407
408
409
410
411
412
413
414
415
416
417
418
419
420
421
422
423
424
425
426
427
428
429
430
431
432
433
434
435
436
437
438
439
440
441
442
443
444
445
446
447
448
449
450
451
452
453
454
455
456
457
458
459
460
461
462
463
464
465
466
467
468
469
470
471
472
473
474
475
476
477
478
479
480
481
482
483
484
485
486
487
488
489
490
491
492
493
494
495
496
497
498
499
500
501
502
503
504
505
506
507
508
509
510
511
512
513
514
515
516
517
518
519
520
521
522
523
524
525
526
527
528
529
530
531
532
533
534
535
536
537
538
539
540
541
542
543
544
545
546
547
548
549
550
551
552
553
554
555
556
557
558
559
560
561
562
563
564
565
566
567
568
569
570
571
572
573
574
575
576
577
578
579
580
581
582
583
584
585
586
587
588
589
590
591
592
593
594
595
596
597
598
599
600
601
602
603
604
605
606
607
608
609
610
611
612
613
614
615
616
617
618
619
620
621
622
623
624
625
626
627
628
629
630
631
632
633
634
635
636
637
638
639
640
641
642
643
644
645
646
647
648
649
650
651
652
653
654
655
656
657
658
659
660
661
662
663
664
665
666
667
668
669
670
671
672
673
674
675
676
677
678
679
680
681
682
683
684
685
686
687
688
689
690
691
692
693
694
695
696
697
698
699
700
701
702
703
704
705
706
707
708
709
710
711
712
713
714
715
716
717
718
719
720
721
722
723
724
725
726
727
728
729
730
731
732
733
734
735
736
737
738
739
740
741
742
743
744
745
746
747
748
749
750
751
752
753
754
755
756
757
758
759
760
761
762
763
764
765
766
767
768
769
770
771
772
773
774
775
776
777
778
779
780
781
782
783
784
785
786
787
788
789
790
791
792
793
794
795
796
797
798
799
800
801
802
803
804
805
806
807
808
809
810
811
812
813
814
815
816
817
818
819
820
821
822
823
824
825
826
827
828
829
830
831
832
833
834
835
836
837
838
839
840
841
842
843
844
845
846
847
848
849
850
851
852
853
854
855
856
857
858
859
860
861
862
863
864
865
866
867
868
869
870
871
872
873
874
875
876
877
878
879
880
881
882
883
884
885
886
887
888
889
890
891
892
893
894
895
896
897
898
899
900
901
902
903
904
905
906
907
908
909
910
911
912
913
914
915
916
917
918
919
920
921
922
923
924
925
926
927
928
929
930
931
932
933
934
935
936
937
938
939
940
941
942
943
944
945
946
947
948
949
950
951
952
953
954
955
956
957
958
959
960
961
962
963
964
965
966
967
968
969
970
971
972
973
974
975
976
977
978
979
980
981
982
983
984
985
986
987
988
989
990
991
992
993
994
995
996
997
998
999
1000

Controlled Copy No. _____

WASHINGTON PUBLIC POWER SUPPLY SYSTEM

WNP-2

PRESSURE/TEMPERATURE LIMITS REPORT

REVISION 0

APRIL 1992

PTLR Implementation

Revision 0 of this report has been reviewed and adopted by the WNP-2 Plant Operating Committee in POC meeting 92-14 held April 1, 1992.

T. M. Erwin, Sr Engineer
Materials & Welding

Date

S. L. Scammon, Supervisor
Plant Technical

Date

S. L. McKay, Manager
Operations

Date

A. G. Hosler, Manager
WNP-2 Licensing

Date

J. W. Baker,
WNP-2 Plant Manager

Date

TABLE OF CONTENTS

| | <u>Page</u> |
|--|-------------|
| 1.0 Pressure/Temperature Limits | 1 |
| 2.0 Allowable Heatup and Cooldown Rates | 1 |
| 3.0 References | 2 |
| Figure 1 WNP-2 Pressure/Temperature Limits | 3 |

1.0 Pressure/Temperature Limits

This Pressure/Temperature Limits Report for WNP-2 has been prepared in accordance with the requirements of Technical Specification Section 6.9.1.12. The pressure/temperature (P/T) limits have been developed using the methodology provided in the identified references.

The following pressure-temperature limits are included in this report.

- 1) Allowable plant heatup and cooldown rates.
- 2) Curve A for hydrostatic or leak testing.
- 3) Curve B for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS.
- 4) Curve C for operation with a critical core other than low power PHYSICS TESTS.

2.0 Allowable Heatup and Cooldown Rates

During system heatup, cooldown, and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the below required heatup and cooldown limits: The pressure and temperature shall be determined at least once every 30 minute interval. The determined temperature and pressure shall be to the right of the appropriate curves in Figure 1.

The following limits shall apply:

1. A maximum heatup of 100° F in any 1-hour period.
2. A maximum cooldown of 100° in any 1-hour period.
3. A maximum temperature change of less than or equal to 20° F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.
4. The reactor vessel flange and head temperature must be greater than or equal to 80° F when reactor vessel head studs are under tension.

3.0 References

1. Supply System calculation ME-02-89-58, Dated 4/2/90.
2. 10 CFR 50 Appendix G requirements.
3. Reg. Guide 1.70 Section 5.3.2/Pressure-Temperature limits

200-100-100

100-100-100

100-100-100

WNP-2 Pressure/Temperature Limits

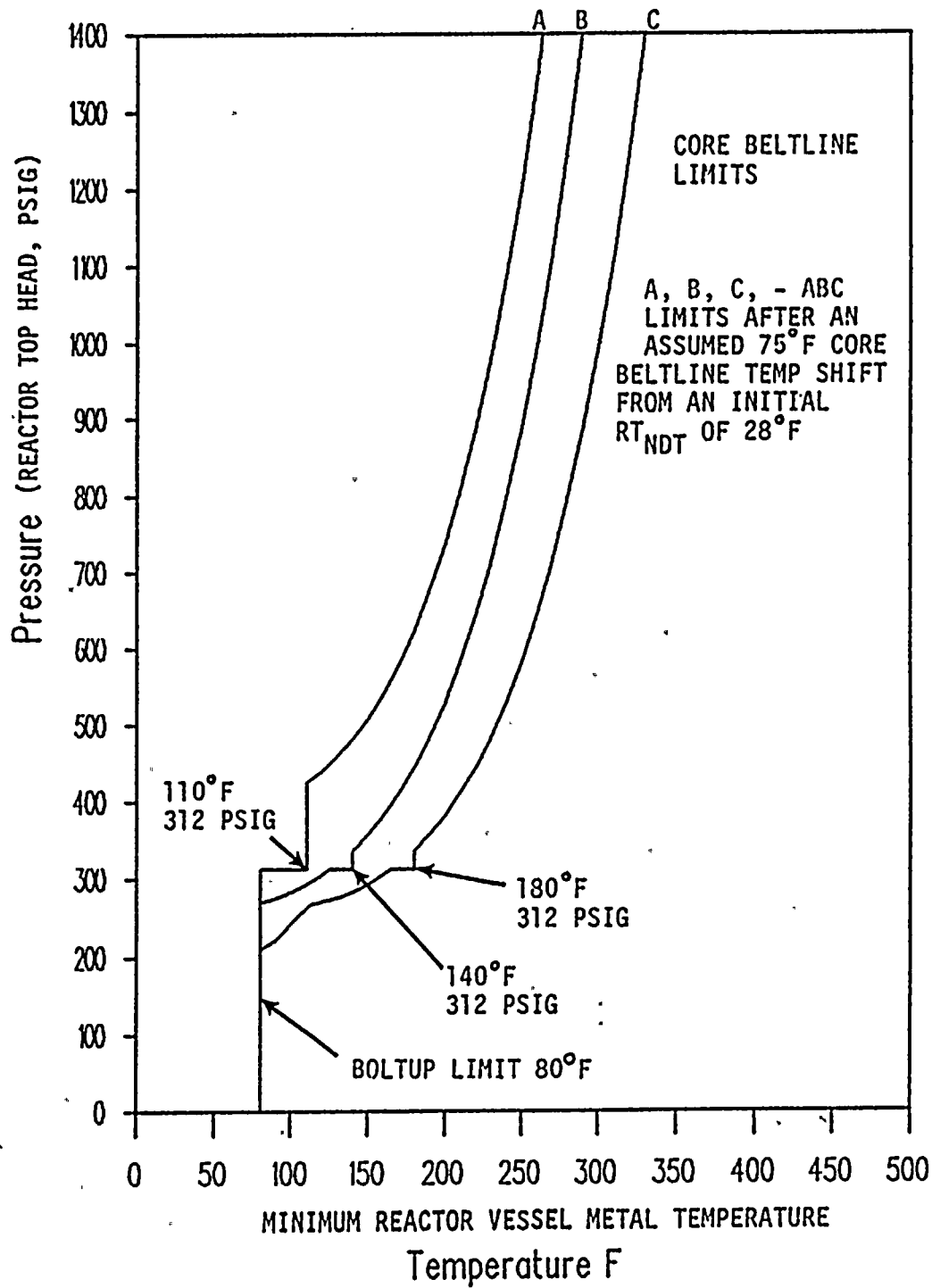


Figure 1

