3.6 LIMITING CONDITION FOR OPERATION

PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the operating status of the reactor coolant system.

Objective:

To assure the integrity and safe operation of the reactor coolant system.

Specification:

A. Thermal Limitations

1. Except as indicated in 3.6.A.2 below, the average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100°F/hr when averaged over a one-hour period.

BWP 3.6.K.1 & 2 LCO

2. A step reduction in reactor coolant temperature of 240°F is permissible so long as the limit in Specification 3.6.A.3 below is met.

3. At all times, the shell flange to shell temperature differential shall not exceed 140°F.

4.6 SURVEILLANCE REQUIREMENT

PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the periodic examination and testing requirements for the reactor coolant system.

Objective:

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification:

A. Thermal Limitations

1. During heatups and cooldowns the following temperatures shall be permanently recorded at 15 minute intervals:

"determined"

30

   a. reactor vessel shell

   b. reactor vessel shell flange

   c. recirculation loops A & B

2. The temperatures listed in 4.6.A.1 shall be permanently recorded subsequent to a heatup or cooldown at 15 minute intervals until three consecutive readings are within 5 degrees of each other.

See Revised BWP 4.6.K2

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

B. Pressurization Temperature

1. The reactor vessel shall be vented and power operation shall not be conducted unless the reactor vessel temperature is equal to or greater than that shown in Curve C of Figure 3.6.1. Operation for hydrostatic or leakage tests, during heatup or cooldown, and with the core critical shall be conducted only when reactor vessel metal temperature is equal to or above that shown in the appropriate curve of Figure 3.6.1. Figure 3.6.1 is effective through 16 effective full power years. At least six months prior to 16 effective full power years new curves will be submitted.

Bup 3.6.K, App.

Bup 3.6.K, LCO

2. The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel shell immediately below the vessel flange is greater than or equal to 80°F.

TSup 3.6.K.4

100

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

Bup 4.6.K

B. Pressurization Temperature

1. Reactor Vessel shell temperature and reactor coolant pressure shall be permanently recorded at 15 minute intervals whenever the shell temperature is below 220°F and the reactor vessel is not vented.

determined

30

Bup 3.6.K, App

Bup 4.6.K.4.6

2. When the reactor vessel head bolting studs are tightened or loosened the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

~~[illegible]~~

FOR INFORMATION ONLY

3.6 LIMITING CONDITION FOR OPERATION
(Cont'd.)

Bwp 3.6.J, LCO

C. Coolant Chemistry

1. a. The reactor coolant activity shall be maintained less than 0.2 microcuries per gram DOSE EQUIVALENT I-131 during Reactor Power operation.

Bwp 3.6.J Mode 1, 2, 3

b. If the reactor coolant activity is greater than 0.2 microcuries per gram and less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131, for more than 48 continuous hours (one continuous time interval) an orderly shutdown shall be immedi-

Bwp 3.6.J, Action 1

4.6 SURVEILLANCE REQUIREMENT
(Cont'd.)

3. Neutron flux monitors and samples shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The monitor and sample program where possible conform to ASTM E 185. The monitors and samples will be removed and tested as outlined in Table 4.6.2 to experimentally verify the calculated values of integrated neutron flux that are used to determine NDTT for Figure 4.6.1.

App H-
Bwp 4.6.K.3

C. Coolant Chemistry

Bwp Table 4.6.J-1

1. a. A sample of reactor coolant shall be taken at least every 96 hours and analyzed for DOSE EQUIVALENT I-131 and total activity content.

Bwp 4.6.J

b. When an isotopic analysis shows reactor coolant activity to be in excess of 0.2 microcuries per gram and less than 4.0 microcuries per gram DOSE EQUIVALENT I-131, additional reactor coolant

Bwp 3.6.J, Actions & 4.6.J

FOR INFORMATION ONLY

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

ately initiated and the unit shall be in cold shutdown within 24 hours.

Hot
12

c. If a sample of reactor coolant activity is greater than 4.0 microcuries per gram DOSE EQUIVALENT I-131, a second sample shall be taken and analyzed within 8 hours. If the second sample indicates a reactor coolant activity greater than 4.0 microcuries per gram DOSE EQUIVALENT I-131, an orderly shutdown shall be initiated and the unit shall be in cold shutdown within 24 hours. Should the second sample indicate a reactor coolant activity less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131, statement 3.6.C.1b shall apply.

Bup 3.6.J, Action 1

Hot
12

Bup 3.6.J, Action 2

2. The reactor coolant water shall not exceed the following limits with steaming rates less than 100,000 pounds per hour except as specified in 3.6.C.3:
Conductivity 2 micro-mho/cm
Chloride ion 0.1 ppm

Bup 3.6.I, LCO

3. For reactor startups the maximum value for conductivity shall not exceed 10 micro-mho/cm and the maximum value for chloride ion concentration shall not exceed 0.1 ppm, for the first 24

TSUP 3.6.I, Actions

0.5

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

TSUP 4.6.J

samples shall be taken and analyzed at least 3 times every 24 hours.

6

c. When reactor coolant activity is greater than 4.0 microcuries per gram DOSE EQUIVALENT I-131, reactor coolant samples shall be taken and analyzed every 8 hours until the reactor is in a cold shutdown condition.

4

TSUP 4.6.J Actions & Table 4.6.J-1

see TSUP 4.6.I.3

2. During startups and at steaming rates below 100,000 pounds per hour, a sample of reactor coolant shall be taken every four hours and analyzed for conductivity and chloride content.

72 per Table 3.6.I-1

3. a. With steaming rates greater than or equal to 100,000 pounds per hour, a reactor coolant sample shall be taken at least every 96 hours and when the continuous conductivity

TSUP 4.6.I.3

4

FOR INFORMATION ONLY

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

BWR 3.6.I, Action

hours after placing the reactor in the power operating condition.

Mode!

BWR 3.6.I, Action

4. Except as specified in 3.6.C.3 above, the reactor coolant water shall not exceed the following limits with steaming rates greater than or equal to 100,000 pounds per hour:
Conductivity 5 micro-mho/cm
10
Chloride ion 0.5 ppm

BWR 3.6.I, Action

5. If Specification 3.6.C.1, 3.6.C.2, 3.6.C.3 or 3.6.C.4 is not met, an orderly shutdown shall be initiated.

D. Coolant Leakage

TS BWR 3.6.H, Applicability

1. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5

TS BWR 3.6.H, LCO

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

BWR 4.6.I.3

monitors indicate abnormal conductivity (other than short-term spikes) and analyzed for conductivity and chloride ion content.

TS BWR 4.6.I.3

b. When the continuous conductivity monitor is inoperable, a reactor coolant sample should be taken at least daily and analyzed for conductivity and chloride ion content.

4 hours

BWR 4.6.H.1

D. Coolant Leakage

1. Reactor coolant system leakage shall be checked by the sump and air sampling system. Sump flow monitoring and recording shall be performed once per 4 hours. Air sampling shall be performed once per day.

8 hrs, not to exceed R

12 hours

BWR 4.6.H.1 & 2

FOR INFORMATION ONLY

3.6 LIMITING CONDITION FOR OPERATION
(Cont'd.)

Bwp 3.6.H.2, LCO

gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm. If these conditions cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

Bwp 3.6.H Actions

2. After completion of the investigation, or containment inspection, specified in 4.6.D.2.a or 4.6.D.2.b, if the leakage is determined to be due to a thru wall pipe crack on the reactor coolant pressure boundary, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

TSwp 3.6.E, Applicability

E. Safety and Relief Valves

1. During reactor power operating conditions and whenever the reactor coolant pressure is greater than 90 psig and temperature

TSwp Modes 1, 2 and 3

4.6 SURVEILLANCE REQUIREMENT
(Cont'd.)

2. The following additional leakage limits shall be met until the recirculation piping indications have been resolved.

   Whenever the reactor is at operating pressure, the following will apply to unidentified leakage:

   a. If a 1 gpm increase over the previous 4 hours occurs or when leakage equals 3 gpm total, an investigation of the cause of the leakage increase will be performed. This investigation should consist of taking drywell air and water samples, and a review of any previous plant evolutions to the extent necessary to determine the source of leakage.

   b. If leakage equals 4 gpm, a containment inspection will be conducted to determine the source of leakage.

E. Safety and Relief Valves

A minimum of 1/2 of all safety valves shall be bench checked or replaced with a bench checked valve each refueling outages.

TSwp 3.6.E LCO

18 mos

FOR INFORMATION ONLY

3.6 LIMITING CONDITION FOR OPERATION
(Cont'd.)

B/d Applic

TS UP 3.6.E LCO

greater than 320°F, all nine of the safety valves shall be operable. The solenoid activated pressure valves shall be operable as required by Specification 3.5.D.

BUP 3/4.5 "ECCS" ADS

TS UP 3.6.E&F Actions

2. If Specification 3.6.E.1 is not met, an orderly shutdown shall be initiated and the reactor coolant pressure and temperature shall be less than or equal to 90 psig and less than or equal to 320° F within 24 hours.

Hot 12 Cold 24

F. Structural Integrity

TS UP 3.6N

The structural integrity of the primary system boundary shall be maintained at the level required by the ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components".

Components of the primary system boundary whose inservice examination reveals the absence of flaw indications or flaw

4.6 SURVEILLANCE REQUIREMENT
(Cont'd.)

BUP 3.6E LCO

The popping point of the safety valves shall be set as follows:

Number of Valves	Set Point (Psig)
1	1135*
2	1240
2	1250
2	1260
2	1260

The allowable set point error for each valve is plus or minus 1%.

18 mos

TS UP 3.6.F LCO 4.6.F

All relief valves shall be checked for set pressure each refueling outage. The set pressures shall be:

Valve No.	Set Point (psig)
203-3A	1124*
203-3B	1101
203-3C	1101
203-3D	1124
203-3E	1124

* Target rock combination safety/relief valve.

The allowable setpoint error for each valve is plus or minus 1%.

F. Structural Integrity

BUP 4.6.N

1. Beginning November 1, 1978, and updated every 40 months thereafter, the component inservice inspection program shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been given by the NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

indications not in excess of the allowable indication standards of this Code are acceptable for continued service. Plant operation with components which have inservice examination flaw indication(s) in excess of the allowable indication standards of the Code shall be subject to NRC approval.

a. Components whose inservice examination reveals flaw indication(s) in excess of the allowable indication standards of the ASME Code, Section XI, are unacceptable for continued service unless the following requirements are met:

TS up 4.0.E

(i) An analysis and evaluation of the detected flaw indication(s) shall be submitted to the NRC that demonstrate that the component structural integrity justifies continued service. The analysis and evaluation shall follow the procedures outlined in Appendix A, "Evaluation of Flaw Indications", of ASME Code, Section XI.

(ii) Prior to the resumption of service, the NRC shall review the analysis and evaluation and

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

either approve resumption of plant operation with the affected component or require that the component be repaired or replaced.

b. For components approved for continued service in accordance with paragraph "a" above, reexamination of the area containing the flaw indication(s) shall be conducted during each scheduled successive inservice inspection. An analysis and evaluation shall be submitted to the NRC following each inservice inspection. The analysis and evaluation shall follow the procedures outlined in Appendix A, "Evaluation of Flaw Indications", of ASME Code, Section XI, and shall reference prior analyses submitted to the NRC to the extent applicable. Prior to resumption of service following each inservice inspection, the NRC shall review the analysis and evaluation and either approve resumption of plant operation with

Bup 4.0.C

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

the affected component or require that the component be repaired or replaced.

c. Repair or replacement of components, including reexaminations, shall conform with the requirements of the ASME Code, Section XI. In the case of repairs, flaws shall be either removed or repaired to the extent necessary to meet the allowable indication standards specified in ASME Code, Section XI.

BWR 4.0 E

BWR 3.6.B LCO

G. Jet Pumps

1. Whenever the Reactor is in the Startup/Hot Standby or Run modes, all jet pumps shall be intact and all operating jet pumps shall be operable. If it is determined that a jet pump is inoperable, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

BWR Mode 1 & 2

BWR 3.6.B Action

HOT

12

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

BWR 4.6.B

G. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the Startup/Hot Standby or Run modes, jet pump integrity and operability shall be checked daily by verifying that the following two conditions do not occur simultaneously:

24 h

a. The recirculation pump flow differs by more than 10% from the established speed-flow characteristics.

BWR 4.6.B.1.a

FOR INFORMATION ONLY

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

TSUP 3.6.B LCO & Actions

2. Flow indication from each of the twenty jet pumps shall be verified prior to initiation of reactor startup from a cold shutdown condition.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

BUP 4.6.B.1.b

b. The indicated total core flow is more than 10% greater than the core flow value derived from established power-core flow relationships.

TSUP 4.6.?.2

2. Additionally, when operating with one recirculation pump with the equalizer valves closed, the diffuser to lower plenum differential pressure shall be checked daily and the differential pressure of any jet pumps in the idle loop shall not vary by more than 10% from established patterns.

indicated flow

TSUP 4.6.B.2

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3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

3. During Dual Loop Operation, the indicated core flow is the sum of the flow indication from each of the twenty jet pumps. During Single Loop Operation (SLO), the indicated core flow must be conservatively adjusted based on station procedures.

e

4. If flow indication failure occurs for two or more jet pumps, immediate corrective action shall be taken. If flow indication for all but one jet pump cannot be obtained within 12 hours an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

TSUP 3.6.B
ACTIONS & LCO

H. Recirculation Pump Flow Limitations

1. Whenever both recirculation pumps are in operation, pump speeds shall be maintained within 10% of each other when power level is greater than 80% and within 15% of each other when power level is less than 80%.

TSUP 3.6.C
LCO

2. If specification 3.6.H.1 cannot be met, one recirculation pump shall be tripped.

TSUP 3.6.C
Action 2

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

3. The baseline data required to evaluate the conditions in Specifications 4.6.G.1 and 4.6.G.2 will be acquired each operating cycle.

TSUP 4.6.C

H. Recirculation Pump Flow Limitations

Recirculation pumps speed shall be checked daily for mismatch.

24 hrs

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3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

3. During Single Loop Operation for more than 24 hours, the following restrictions are required:

TSup 3.6.A, Action 1

a. The recirculation pump in the idle loop shall be electrically prohibited from starting except to permit testing in preparation for returning to service.

TSup 3.6.A, Action 1.e

b. The flow biased RBM Rod Block LSSS shall be reduced by 4.0% (Specification 3.2.C.1);

TSup 3.6.A, Action 1.c

c. The flow biased APRM Rod Block LSSS shall be reduced by 3.5% (Specification 2.1.B);

TSup 3.6.A, Action 1.c

d. The flow biased APRM scram LSSS shall be reduced by 3.5% (Specification 2.1.A.1);

TSup 3.6.A, Action 1.c

e. The MCPR Safety Limit shall be increased by 0.01 (Specification 1.1.A);

TSup 3.6.A, Action 1.a

f. The rated flow MCPR Operating Limit shall be increased by 0.01 (Specification 3.5.L.2);

TSup 3.6.A, Action 1.b

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

3. Deleted

LIMITING CONDITION FOR OPERATION
(Cont'd.)

g. The MAPLHGR Operating Limit shall be reduced by the appropriate multiplicative factor from the Core Operating Limits Report (Specification 3.5.I). If, concurrently, one Automatic Pressure Relief Subsystem relief valve is out-of-service, the MAPLHGR Operating Limit shall be reduced by the appropriate multiplicative factor from the Core Operating Limits Report.

TSUP 3.6.A, Action 1.d

TSUP 3.6.A, Action 2

4. With no reactor coolant system recirculation loops in operation, reduce core thermal power to less than 25% of rated within 2 hours and place the unit in hot shutdown within the following 12 hours.

Startup in 6 hrs

6

5. Idle Recirculation Loop Startup

An idle recirculation pump shall not be started unless the temperature differential between the reactor vessel steam space coolant and the bottom head drain line coolant is less than or equal to 145°F*, and:

TSUP 3.6.D LCO

a. When both pumps have been idle, unless the temperature differential between the reactor coolant within the idle loop to be started up and the coolant in the reactor pressure vessel is less than or equal to 50°F, or

TSUP 3.6.D.1 LCO

b. When only one loop has been idle, unless the temperature differential between the reactor coolant within the idle and operating recirculation loops is less than or equal to 50°F and the speed of the operating pump is less than or equal to 43% of rated pump speed.

TSUP 3.6.D.2 LCO

I. Snubbers (Shock Suppressors)

4.6 SURVEILLANCE REQUIREMENT
(Cont'd.)

5. Idle Recirculation Loop Startup

The temperature differentials and flow rates shall be determined to be within the limits within 15 minutes prior to startup of an idle recirculation loop.

TSUP 4.6.D

TSUP 3/4.8.F

I. Snubbers (Shock Suppressors)

The following surveillance requirements apply to safety related snubbers.

*Only applicable with reactor pressure vessel steam space pressure ≥ 25 psig.

TSUP 3.6.D, footnote (a)

3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

1. During all modes of operation except cold shutdown and refuel, all safety related snubbers shall be operable except as noted in Specification 3.6.I.2 through 3.6.I.4.

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TSUP 3/4.8.F

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

1. Visual Inspection

An independent visual inspection shall be performed on the safety related hydraulic and mechanical snubbers in accordance with the schedule below.

a. All hydraulic snubbers whose seal material has been demonstrated by operating experience, lab testing or analysis to be compatible with the operating environment shall be visually inspected. This inspection shall include, but not necessarily be limited to, inspection of the hydraulic fluid reservoir, fluid connections, and linkage connection to the piping and anchor to verify snubber operability.

b. All mechanical snubbers shall be visually inspected. This inspection shall consist of, but not necessarily be limited to, inspection of the snubber and attachments to the piping and anchor for indications of damage or impaired operability.

3.6 LIMITING CONDITION FOR OPERATION
(Cont'd.)

2. From and after the time a snubber is determined to be inoperable, continued reactor operation is permissible only during the succeeding 72 hours unless the snubber is sooner made operable or replaced.

4.6 SURVEILLANCE REQUIREMENT
(Cont'd.)

No. of Snubbers Found Inoperable During Inspection Interval	Next Required Inspection Interval
0	18 months plus or minus 25%
1	12 months plus or minus 25%
2	6 months plus or minus 25%
3,4	124 days plus or minus 25%
5,6,7	62 days plus or minus 25%
8 or more	31 days plus or minus 25%

The required inspection interval shall not be lengthened more than one step at a time.

Snubbers may be categorized in two groups, "accessible" or "inaccessible," based on their accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

2. Functional Testing

a. Once each refueling cycle, a representative sample of approximately 10% of the hydraulic snubbers shall be functionally tested for operability, including:

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BUP 3/4.8.F

FOR INFORMATION ONLY

3.6 LIMITING CONDITION FOR OPERATION
(Cont'd.)

4.6 SURVEILLANCE REQUIREMENT
(Cont'd.)

(i) Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

(ii) Snubber bleed, or release rate, where required, is within the specified range in compression or tension.

For each unit and subsequent unit found inoperable, an additional 10% of the hydraulic snubbers shall be tested until no more failures are found or all units have been tested.

b. Once each refueling cycle, a representative sample of approximately 10% of the mechanical snubbers shall be functionally tested for operability. The test shall consist of two parts:

BUP
3/4.8.F

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3.6 LIMITING CONDITION FOR OPERATION
(Cont'd.)

75up
3/4.8.F

4.6 SURVEILLANCE REQUIREMENT
(Cont'd.)

(i) Verification that the force that initiates free movement of the snubber in either tension or compression is less than the specified maximum breakaway friction force.

(ii) Verify that the activation (restraining action) is achieved within the specified range of acceleration or velocity, as applicable based on snubber design in both tension and compression.

For each unit and subsequent unit found inoperable, an additional 10% of the mechanical snubbers shall be so tested until no more failures are found or all units have been tested.

c. In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the resampling.

3.6 LIMITING CONDITION FOR OPERATION
(Cont'd.)

3. If the requirements of 3.6.I.1 and 3.6.I.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in cold shutdown or refuel condition within 36 hours.

4. If a snubber is determined to be inoperable while the reactor is in the cold shutdown or refuel mode, the snubber shall be made operable or replaced prior to reactor startup.

4.6 SURVEILLANCE REQUIREMENT
(Cont'd.)

3. When a snubber is deemed inoperable, a review of all pertinent facts shall be conducted to determine the snubber mode of failure and to decide if an engineering evaluation should be performed on the supported system or components. If said evaluation is deemed necessary, it will determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

4. If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen in place, the cause will be evaluated and, if determined to be a generic deficiency, all snubbers of the same design subject to the same defect shall be functionally tested.

TSUP 3/4.8.F

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3.6 LIMITING CONDITION FOR OPERATION (Cont'd.)

5. Snubbers may be added or removed from safety related systems without prior license amendment.

4.6 SURVEILLANCE REQUIREMENT (Cont'd.)

5. Snubber service life monitoring shall be followed by existing station record systems, including the central filing system, maintenance files, safety related work packages, and snubber inspection records. The above record retention methods shall be used to prevent the hydraulic snubbers from exceeding a service life of 10 years and the mechanical snubbers from exceeding a service life of 40 years (lifetime of the plant).

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3/4.8.F

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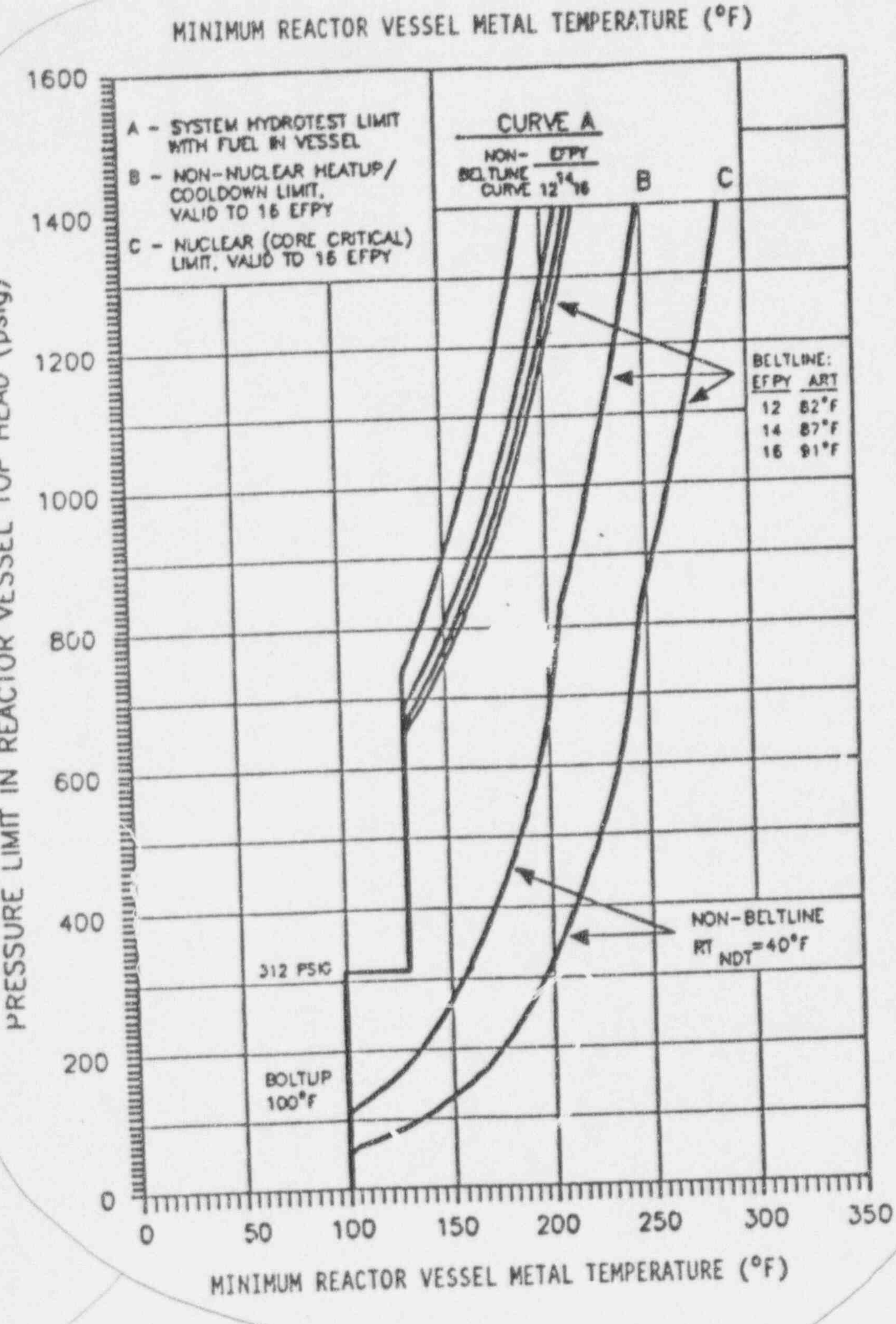

FIGURE 3.6.1.

TSUP
FIGURE 3.6.K-1

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QUAD-CITIES

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3.6/4.6 PRIMARY SYSTEM BOUNDARY

LIMITING CONDITIONS FOR OPERATION

Applicability:

Applies to the operating status of the reactor coolant system.

Objective:

To assure the integrity and safe operation of the reactor coolant system.

SURVEILLANCE REQUIREMENTS

Applicability:

Applies to the periodic examination and testing requirements for the reactor coolant system.

Objective:

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

SPECIFICATIONS

3.6 (Limiting Conditions for Operation)

A. Thermal Limitations

1. Except as indicated in Specification 3.6.A.2 below, the average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100°F/hr when averaged over a 1-hour period.

maximum

TSUP 3.6.K.1 & 2 LCO

2. A step reduction in reactor coolant temperature of 240°F is permissible so long as the limit in Specification 3.6.A.3 below is met.

3. At all times, the shell flange to shell temperature differential shall not exceed 140°F.

4. The recirculation pump in an idle recirculation loop shall not be started unless the coolant in that loop is within 50°F of the operating loop coolant temperature.

TSUP 3.6.D.2 LCO

4.6 (Surveillance Requirements)

A. Thermal Limitations

TSUP 4.6.K.1

1. During heatups and cooldowns the following temperatures shall be permanently recorded at 15-minute intervals:

determined

see

a. reactor vessel shell,

b. reactor vessel shell flange, and

c. recirculation loops A and B.

TSUP 4.6.K.1, 2 & 3

2. The temperatures listed in Specification 4.6.A.1 shall be permanently recorded subsequent to a heatup or cooldown at 15-minute intervals until three consecutive readings at each given location are within 5 degrees of each other.

See Revised TSUP 4.6.K.2

TSUP 4.6.K

B. Pressurization Temperature

1. The reactor vessel shall be vented and power operation shall not be conducted unless the reactor vessel temperature is equal to or greater than that shown in Figure 3.6-1.

Operation for hydrostatic or leakage tests (Curve A), during heatup or cooldown (Curve B), or with the core critical (Curve C) shall be conducted only when the reactor vessel temperature is equal to or above that shown in the appropriate curve of Figure 3.6-1. Figure 3.6-1 is effective through 16 EFPY. At least six months prior to 16 EFPY new curves will be submitted.

2. The reactor vessel head bolting studs shall not be under tension unless the temperature of the vessel shell immediately below the vessel flange is ≥ 100°F.

C. Coolant Chemistry

1. The steady-state radioiodine concentration in the reactor coolant shall not exceed 5 µCi of I-131 dose equivalent per gram of water.

0.2

TSUP 3.6.J, LCO

B. Pressurization Temperature

1. Reactor vessel shell temperature and reactor coolant pressure shall be permanently recorded at 15-minute intervals whenever the shell temperature is below 220°F and the reactor vessel is not vented.

30

2. Neutron flux monitors and samples shall be installed in the reactor vessel adjacent to the vessel wall at the core midplane level. The monitor and sample program shall conform to ASTM E 185-66. The monitors and samples shall be removed and tested in accordance with the guidelines set forth in 10CFR50 Appendix H to experimentally verify the calculated values of integrated neutron flux that are used to determine the NDTT for Figure 3.6-1.

TSUP 4.6.K.3

3. When the reactor vessel head bolting studs are tightened or loosened, the reactor vessel shell temperature immediately below the head flange shall be permanently recorded.

TSUP 4.6.K.4.b

C. Coolant Chemistry

1. a. A sample of reactor coolant shall be taken at least every 96 hours and analyzed for radioactive iodines of I-131 through I-135 during power operation. In addition, when chimney monitors indicate an increase in radioactive gaseous effluents of 25% or 5000 µCi/sec. whichever is greater, during steady-state reactor operation, a reactor coolant sample shall be taken and analyzed for radioactive iodines.

75

100,000

TSUP 4.6.J, Table 4.6.J-1

b. An isotopic analysis of a reactor coolant sample shall be made at least once per month.

TSUP 4.6.J Table 4.6.J-1

c. Whenever the steady-state radioiodine concentration of prior operation is greater than 1% but less

TSUP 3.6.J, Action

TS(w) 3.6.J, Actions

than 10% of Specification 3.6.C.1, a sample of reactor coolant shall be taken within 24 hours of any reactor startup and analyzed for radioactive iodines of I-131 through I-135.

2. The reactor coolant water shall not exceed the following limits with steaming rates less than 100,000 lb/hr except as specified in Specification 3.6.C.3:

TS up 3.6.I, Applicability

mode dependent limits

TS up 3.6.I, LCO

conductivity 2 μmho/cm

chloride ion 0.1 ppm

3. For reactor startups, the maximum value for conductivity shall not exceed 10 μmho/cm, and the maximum value for chloride ion concentration shall not exceed 0.1 ppm for the first 24 hours after placing the reactor in the power operating condition.

TS up 3.6.I, Actions

0.3

TS up Mode 1

4. Except as specified in Specification 3.6.C.3 above, the reactor coolant water shall not exceed the following limits with steaming rates greater than or equal to 100,000 lb/hr:

TS up 3.6.I, Actions

conductivity 10 μmho/cm

chloride ion 1.0 ppm

5. If Specification 3.6.C.1, 3.6.C.2, 3.6.C.3, or 3.6.C.4 is not met, an orderly shutdown shall be initiated.

TS up 3.6.I, Actions

TS up 3.6.H Applicability

D. Coolant Leakage

1. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212° F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm.

TS up 3.6.H.3 LCO

TS up 3.6.H.2 LCO

through I-135.

TS up 3.6.J, Actions

d. Whenever the steady-state radioiodine concentration of prior operation is greater than 10% of Specification 3.6.C.1, a sample of reactor coolant shall be taken prior to any reactor startup and analyzed for radioactive iodines of I-131 through I-135 as well as the coolant sample and analyses required by Specification 4.6.C.1.c above.

2. During startups and at steaming rates below 100,000 lb/hr, a sample of reactor coolant shall be taken every 4 hours and analyzed for conductivity and chloride content.

72 hrs or per Table 3.6.I-1

TS up 4.6.I.2, 4.6.I.3

See TS up 4.6.I.3

3. a. With steaming rates greater than or equal to 100,000 lb/hr, a reactor coolant sample shall be taken at least every 96 hours and when the continuous conductivity monitors indicate abnormal conductivity (other than short-term spikes) and analyzed for conductivity and chloride ion content.

(9)

TS up 4.6.I.3

b. When the continuous conductivity monitor is inoperable, a reactor coolant sample should be taken at least daily and analyzed for conductivity and chloride ion content.

TS up 4.6.I.3

every 4 hours

D. Coolant Leakage

TS up 4.6.H.1

Reactor coolant system leakage shall be checked by the sump and air sampling system. Sump flow monitoring and recording shall be performed once per shift. Air sampling shall be performed once per day.

TS up 4.6.H.1, 4.6.H.2

8 hours, not to exceed 12 hours

12 hours

2. Both the sump and air sampling systems shall be operable during reactor power operation. From and after the date that one of these systems is made or found to be inoperable for any reason, reactor power operation is permissible only during the succeeding 7 days.

3. If the conditions in 1 or 2 above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

TSUP 3.6.H, Actions

Hot 12, Cold 24

TSUP 3.6.E, Notes 1,2,3

E. Safety and Relief Valves

1. Prior to reactor startup for power operation, during reactor power operating conditions, and whenever the reactor coolant pressure is greater than 90 psig and temperature greater than 320° F, all nine of the safety valves shall be operable. The solenoid-activated pressure valves shall be operable as required by Specification 3.5.D.

TSUP 3.6.E, Applicability

TSUP 3.6.E, LCO

2. If Specification 3.6.E.1 is not met, the reactor shall remain shut down until the condition is corrected or, if in operation, an orderly shutdown shall be initiated and the reactor coolant pressure and temperature shall be below 90 psig and 320° F within 24 hours.

TS 3/4.5, ECCS

TSUP 3.6.E Action Hot-12, Cold-24

F. Structural Integrity

The structural integrity of the primary system boundary shall be maintained at the level required by the ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components", 1974 Edition, Summer 1975 Addenda (ASME Code Section XI).

TSUP 3.6.N

TSUP 4.0.E defines

TSUP 3.6.E, LCO

18 month

E. Safety and Relief Valves

A minimum of 1/2 of all safety valves shall be bench checked or replaced with a bench checked valve each refueling outage. The popping point of the safety valves shall be set as follows:

Number of Valves	*Setpoint (psig)*
1	1135[1]
2	1240
2	1250
4	1260

The allowable setpoint error for each valve is ± 1%.

All relief valves shall be checked for set pressure each refueling outage. The set pressures shall be:

18 month

TSUP 4.6.F & 3.6.F, LCO

Number of Valves	*Setpoint (psig)*
1	≤1135[1]
2	≤1115
2	≤1135

[1]Target Rock combination safety/relief valve.

F. Structural Integrity

The nondestructive inspections listed in Table 4.6-1 shall be performed as specified in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1971 Edition, Summer 1971 Addenda. The results obtained from compliance with this specification will be evaluated after 5 years and the conclusions will be reviewed with the NRC.

TSUP 4.6.N & 4.0.E

Components of the primary system boundary whose inservice examination reveals the absence of flaw indications or flaw indications not in excess of the allowable indication standards of this Code are acceptable for continued service. Plant operation with components which have inservice examination flaw indication(s) in excess of the allowable indication standards of the Code shall be subject to NRC approval.

1. Components whose inservice examination reveals flaw indication(s) in excess of the allowable indication standards of the ASME Code, Section XI, are unacceptable for continued service unless the following requirements are met:

   a. An analysis and evaluation of the detected flaw indication(s) shall be submitted to the NRC that demonstrate that the component structural integrity justifies continued service. The analysis and evaluation shall follow the procedures outlined in Appendix A, "Evaluation of Flaw Indications", of ASME Code, Section XI.

   b. Prior to the resumption of service, the NRC shall review the analysis and evaluation and either approve resumption of plant operation with the affected component or require that the component be repaired or replaced.

2. For components approved for continued service in accordance with paragraph 1, reexamination of the area containing the flaw indication(s) shall be conducted during each scheduled successive inservice inspection. An analysis and evaluation shall be submitted to the NRC following each inservice inspection. The analysis and evaluation shall follow the procedures outlined in Appendix A, "Evaluation of Flaw Indications", of ASME Code, Section XI, and shall reference prior analyses submitted to the NRC to the extent applicable. Prior to resumption of service following each inservice inspection, the NRC shall

TSUP
4.0.E

review the analysis and evaluation and either approve resumption of plant operation with the affected component or require that the component be repaired or replaced.

3. Repair or replacement of components, including reexaminations, shall conform with the requirements of the ASME Code, Section XI. In the case of repairs, flaws shall be either removed or repaired to the extent necessary to meet the allowable indication standards specified in ASME Code, Section XI.

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TSUP 4.6.B

G. Jet Pumps

Mode 1,2

OPERABLE

1. Whenever the reactor is in the Startup/Hot Standby or Run modes, all jet pumps shall be intact, and all operating jet pumps shall be operable. If it is determined that a jet pump is inoperable, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

HOT

12

TSUP 3.6.B LCO & ACTIONS

TSUP 3.6.B. LCO, ACTIONS

Footnote (a)

18

2. Flow indication from 19 of the 20 jet pumps shall be verified prior to initiation of reactor startup from a cold shutdown condition.

indicated flow

3. The indicated core flow is the sum of the flow indication from each jet pump with operable flow indication. In addition, for any jet pump with inoperable flow indication, the flow indication from the companion jet pump on the same jet pump riser shall be summed a second time to compensate for the flow through the jet pump with inoperable flow indication. If flow indication failure occurs for three or

G. Jet Pumps

Mode 1,2

24 hrs

1. Whenever there is recirculation flow with the reactor in the Startup/Hot Standby or Run modes, jet pump integrity and operability shall be checked daily by verifying that two of the following conditions do not occur simultaneously:

TSUP 4.6.B1.a

a. The recirculation pump flow differs by more than 10% from the established speed-flow characteristics.

TSUP 4.6.B1.b

b. The indicated total core flow is more than 10% greater than the core flow value derived from established core plate DP/core flow relationships.

TSUP 4.6.B1.c

c. Individual jet pump flow for any jet pump differs by more than 10% from established flow to average loop jet pump flow characteristics.

TSUP 4.6.B.2

2. Additionally, when operating with one recirculation pump with the equalizer valves closed, the diffuser to lower plenum differential pressure shall be checked daily, and the differential pressure of any jet pump in the idle loop shall not vary by more than 10% from established patterns.

TSUP 4.6.B.2.c

3. The baseline data required to evaluate the conditions in Specifications 4.6.G.1 and 4.6.G.2 will be acquired each operating cycle.

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more jet pumps, immediate corrective action shall be taken. If flow indication for all but two jet pumps cannot be obtained within 12 hours, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

TS ûf 3.6.B Action 2

Hot

12

4. If flow indication failure occurs for both jet pumps on the same jet pump riser, immediate corrective action shall be taken. If flow indication for at least one of the jet pumps cannot be obtained within 12 hours, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

TS ûf 3.6.B, Action 3

Hot

12

5. If flow indication failure occurs for both calibrated (double-tap) jet pumps on the same recirculation loop, immediate corrective action shall be taken. If flow indication for at least one of the jet pumps cannot be obtained within 12 hours, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

TS ûf 3.6.B Action 4

Hot

12

H. Recirculation Pump Flow Limitations

1. Whenever both recirculation pumps are in operation, pump speeds shall be maintained within 10% of each other when power level is greater than 80% and within 15% of each other when power level is less than 80%.

TS ûf 3.6.C LCO

2. If Specification 3.6.H.1 cannot be met, one recirculation pump shall be tripped.

TS ûf 3.6.C, ACTION 2

H. Recirculation Pump Flow Limitations

Recirculation pumps speed shall be checked daily for mismatch.

every 24 hrs

TS ûf 4.6.C

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3. During Single Loop Operation for more than 24 hours, the following restrictions are required:

TSUP 3.6.A, Action 1

   a. The MCPR Safety Limit shall be increased by 0.01 (T.S. 1.1A);

   TSUP 3.6.A, Action 1.a

   b. The MCPR Operating Limit, as specified in the CORE OPERATING LIMITS REPORT, shall be increased by 0.01 (T.S. 3.5.K);

   TSUP 3.6.A, Action 1.b

   c. The flow biased APRM Scram and Rod Block Setpoints shall be reduced by 3.5% to read as follows:

      T.S. 2.1.A.1;
      S $\leq$ .58 WD + 58.5

      T.S. 2.1.A.1;*
      S $\leq$ (.58 WD + 58.5) FRP/MFLPD

      T.S 2.1.B;
      S $\leq$ .58 WD + 46.5

      T.S. 2.1.B; *
      S $\leq$ (.58 WD + 46.5) FRP/MFLPD

      T.S. 3.2.C (Table 2.1-3);*
      APRM upscale $\leq$ (.58 WD + 46.5) FRP/MFLPD

   * In the event that MFLPD exceeds FRP.

   TSUP 3.6.A, Action 1.c

   d. The flow biased RBM Rod Block setpoints, as specified in the CORE OPERATING LIMITS REPORT, shall be reduced by 4.0%.

   TSUP 3.6.A, Action 1.c

   e. The recirculation pump in the idle loop shall be electrically prohibited from starting except to permit testing in preparation for returning to service.

   TSUP 3.6.A, Action 1.e

TSUP 3.6.A, Action 2

4. With no reactor coolant system recirculation loops in operation, reduce core thermal power to less then 25% of rated within 2 hours and place the unit in hot shutdown within the following 12 hours.

startup in 6 hours

6

5. Idle Recirculation Loop Startup

An idle recirculation pump shall not be started unless the temperature differential between the reactor vessel steam space coolant and the bottom head drain line coolant is less than or equal to 145°F*, and:

TSUP 3.6.D, LCO

a. When both pumps have been idle, unless the temperature differential between the reactor coolant within the idle loop to be started up and the coolant in the reactor pressure vessel is less than or equal to 50°F, or

TSUP 3.6.D.1 LCO

b. When only one loop has been idle, unless the temperature differential between the reactor coolant within the idle and operating recirculation loops is less than or equal to 50°F and the speed of the operating pumps is less than or equal to 45% of rated pump speed.

TSUP 3.6.D.2 LCO

5. Idle Recirculation Loop Startup

The temperature differentials and flow rates shall be determined to be within the limits within 15 minutes prior to startup of an idle recirculation loop.

TSUP 4.6.D

*Only applicable with reactor pressure vessel steam space pressure ≥ 25 psig.

TSUP 3.6.D, footnote (a)

I. Shock Suppressors (Snubbers)

1. During all modes of operation except Shutdown and Refuel, all snubbers on safety related piping systems shall be operable except as noted in 3.6.I.2 following.

2. From and after the time that a snubber is determined to be inoperable, continued reactor operation is permissible during the succeeding 72 hours only if the snubber is sooner made operable.

3. If the requirements of 3.6.I.1 and 3.6.I.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 36 hours.

4. If a snubber is determined to be inoperable while the reactor is in the Shutdown or Refuel mode, the snubber shall be made operable prior to reactor startup.

I. Shock Suppressors (Snubbers)

The following surveillance requirements apply to all snubbers on safety related piping systems.

1. Visual inspections shall be performed in accordance with the following schedule utilizing the acceptance criteria given by Specification 4.6.I.2.

Number of Snubbers Found Inoperable During Inspection or During Inspection Interval	Next Required Inspection Interval
0	18 months ±25%
1	12 months ±25%
2	6 months ±25%
3,4	124 days ±25%
5,6,7	62 days ±25%
≥8	31 days ±25%

The required inspection interval shall not be lengthened more than one step at a time.

Snubbers may be categorized in two groups, 'accessible' or 'inaccessible' based on their accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

RELOCATED
TSUP 3/4.8.F

Snubber service life monitoring shall be followed by the snubber surveillance inspection records and maintenance history records. The above record retention method shall be used to prevent the snubbers from exceeding a service life.

2. Visual inspections shall verify:

   a. There are no visible indications of damage or impaired operability, and

   b. Attachments to the foundation or supporting structure are secure.

3. Once each refueling cycle a representative sample of 10% of the total of each type of snubber in use in the plant shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test criteria, an additional 10% of that type of snubber shall be functionally tested.

4. The mechanical snubber functional tests shall verify:

   a. That the breakaway force that initiates free movement of the snubber rod in either tension or compression is less than the specified maximum force.

   b. That the activation (restraining action) is achieved within the specified range of acceleration in both tension and compression.

TS up 3/4.8.F

5. When a snubber is deemed inoperable, a review shall be conducted to determine the mode of failure and to decide if an engineering evaluation should be performed. If the engineering evaluation is deemed necessary, it will determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

6. If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if determined to be generically deficient all snubbers of the same design, subject to the same defect shall be functionally tested.

7. In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the resampling.

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3/4.8.F

QUAD CITIES

DPR-30

PRESSURE LIMIT AS A FUNCTION OF VESSEL METAL TEMPERATURE

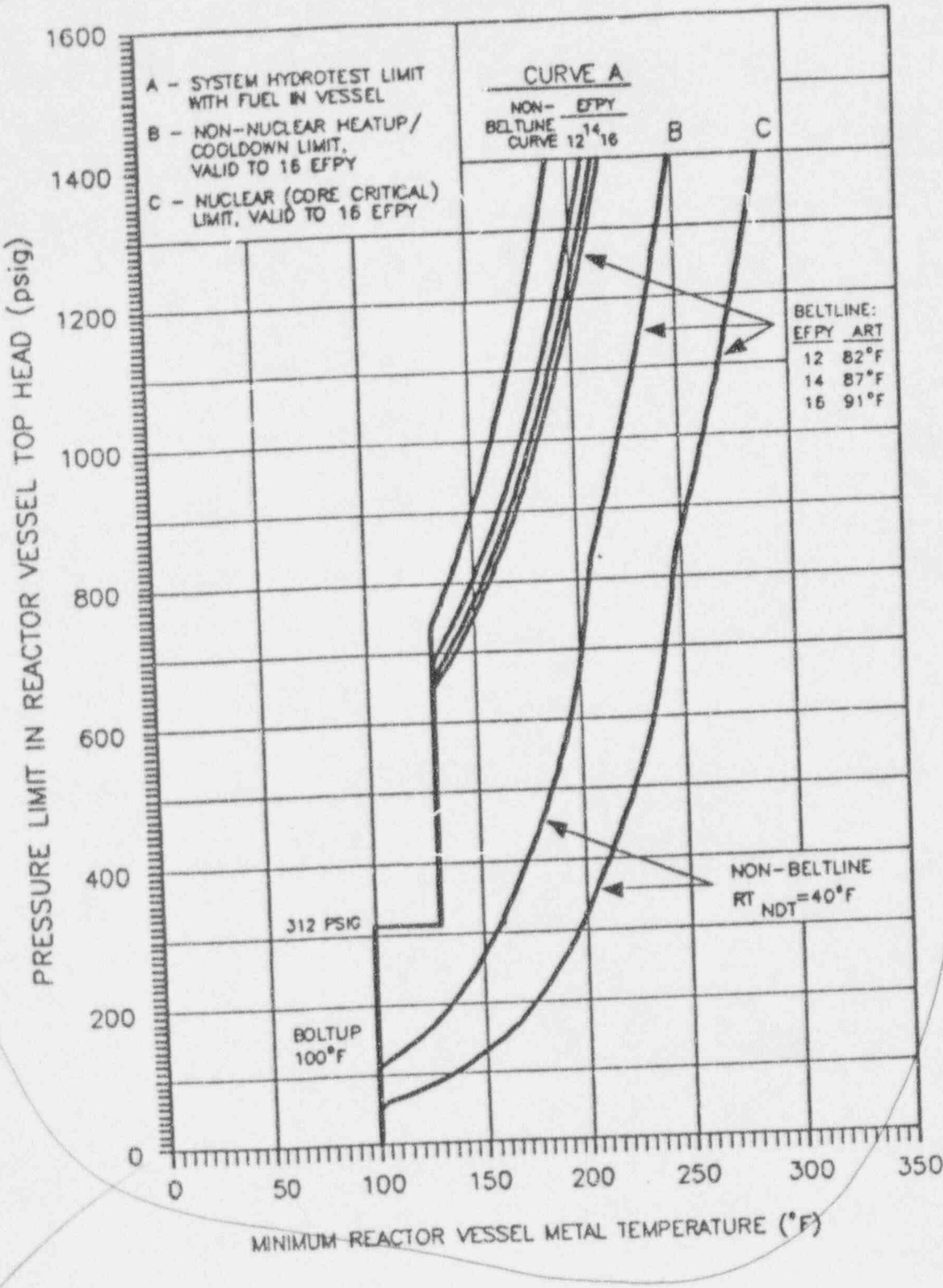

FIGURE 3.6-1

BNF
FIGURE 3.6.K-1

TABLE 4.6-1

INSERVICE INSPECTION REQUIREMENTS FOR QUAD-CITIES

Category	Component Parts to be Examined	Examination Method	Frequency of Examination	Examinations[1]
A	Longitudinal and circumferential shell welds in core region			Note: Not applicable with present plant design
B	Longitudinal and circumferential welds in shell (other than those of categories A and C) and meridional and circumferential seam welds in bottom head and closure head (other than those of Category C)	Volumetric	During each 10-year inspection interval (for 10% of each longitudinal and meridional 5% circumferential length seam)	Accessible top 10 feet of vertical vessel weld in two places (100% inspected in 10 years for approximately 2 feet each refueling outage) 10% of meridional seam welds in vessel closure head and 5% of circumferential welds in vessel closure head Note: Bottom head closure not applicable with present plant design
C	Vessel-to-flange and head-to-flange circumferential welds	Volumetric	Cumulative 100% coverage at end of 10-year interval	Equivalent to 10% of vessel-to-flange and head-to-flange circumferential weld are a each refueling outage
D	Primary nozzle-to-vessel and nozzle-to-head welds and nozzle-to-vessel, nozzle-to-head inside radiused section	Volumetric	Cumulative 100% coverage at end of 10-year interval	Nozzle welds: Recirculation outlet[2]: once every 5 years Recirculation inlet[10]: at least once each refueling outage Core spray inlet[2]: once every 5 years Control rod drive return[1]: once every 10 years Standby liquid control[1]: once every 10 years Head instrumentation[2]: once every 5 years Head spray inlet[1]: once every 10 years

B&P
4.0.E

TABLE 4.6-1 (Cont'd)

Category	Component Parts to be Examined	Examination Method	Frequency of Examination	Examinations[1]
E	Partial penetration welds including control rod drive penetrations and vessel instrumentation nozzles	Visual	The examinations performed during each inspection interval shall cover at least 25% of each group of penetrations of comparable size and function	The area surrounding each penetration shall be examined for evidence of leakage during pressure testing
F	Primary nozzles to safe-end welds	Visual, surface, and volumetric	Cumulative 100% coverage at end of 10-year interval	Safe-ended nozzles: Recirculation outlet[2]: once every 5 years Recirculation inlet[10]: at once each refueling outage Core spray inlet[2]: once every 5 years Control rod drive[1]: once every 10 years Standby liquid control[1]: once every 10 years Head instrumentation[2]: once every 5 years Head spray inlet[1]: once every 10 years
G-1	Closure studs and nuts	Volumetric and visual or surface	Cumulative 100% coverage at end of 10-year interval	100% of vessel studs and nuts will be inspected each refueling outage
	Ligaments between threaded stud holes	Volumetric	Cumulative 100% coverage at end of 10-year interval	Equivalent to 10% of ligaments each refueling outage. Examination of bushings, threads, and ligaments in base material of flanges may be performed from the face of the flange and are required to be examined only when the connection is disassembled.
	Closure washers, bushings	Visual	Cumulative 100% coverage at end of 10-year interval	Equivalent to 10% of washers each refueling outage, bushings not applicable with present design.
	Pressure-retaining bolting ≥2 inch diameter	Visual and volumetric	Cumulative 100% coverage at end of 10-year interval	Equivalent to 10% of recirculating pump bolts each refueling outage.

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TABLE 4.6-1 (Cont'd)

Category	Component Parts to be Examined	Examination Method	Frequency of Examination	Examinations[1]
G-2	Pressure-retaining bolting <2 inch diameter	Visual	Cumulative 100% of coverage at end of 10-year interval	Bolting will be examined when bolting is removed or when the bolted connected is broken or disassembled. For bolting which is not removed or where the bolted connection is not broken, the inspection will consist of a visual examination to detect signs of distress or evidence of leaking.
H	Integrally welded vessel supports	Volumetric	During 10-year interval	10% (approximately 8 ft) of lineal feet of vessel support skirt welding in 10th year.
I	Closure head cladding	Visual and surface or volumetric	During 10-year interval	During the 10-year interval, at least six patches (each 36 in^2) evenly distributed in the closure head.
	Vessel cladding	Visual	During 10-year interval	6 patches (each 36 in^2) evenly distributed in the accessible sections of the vessel shell shall be examined.

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TABLE 4.6-1 (Cont'd)

Category	Component Parts to be Examined	Examination Method	Frequency of Examination	Examinations[1]		
				System	Pipe Sizes	Unit 2 Total Welds
J -	Circumferential and longitudinal pipe welds (Refer to Note 2 at the end of this table for a breakdown of these welds.)	Visual and volumetric	Cumulative 25% of all weld joints (selectively distributed among the higher stress joints in entire system) every 10 years. Group I and Group II welds (See Note 1 for location breakdown) on main feedlines and main steamlines shall be inspected in 10 years during the first period. At least 25% of the welds shall be inspected at approximately each 2-1/2-year interval. Group I welds shall be inspected during each 10-year period thereafter.	Shutdown cooling	20-in.	17
				RCIC	3-in.,4-in.	33
				Reactor water cleanup	4-in.,6-in.	27
				CRD hydraulic system	3-in.,4-in.	18
				RHR	16-in.	29
				Head spray	4-in.	28
				Core spray piping	10-in.	32
				HPCI	10-in., 14-in.	24
				Feed piping	4-in.,12-in. 18-in.	96
				Recirculation	4-in.,12-in.	
				Main Steam	22-in., 22-in., 28-in.	135
					3-in.,20-in.	120
K-1	Integrally-welded external support attachments for piping, valves, and pumps	Visual and volumetric	100% cumulative in first 10 years 25% cumulative in each following 10-year inspection interval	Welds to the pressure-containing boundary, the base metal beneath the weld zone, and along the support attachment member for a distance of two base metal thicknesses.		
K-2	Support members and structures for piping, valves, and pumps whose structural integrity is relied upon to withstand design loads and seismic-induced displacements.	Visual	100% cumulative during each 10-year inspection interval	Support settings of constant and variable spring type hangers, snubbers, and shock absorbers shall be inspected to verify proper distribution of design loads among the associated support components.		
L-1	Pump casing welds	Visual and volumetric	One pump of each type during 10-year interval	Not applicable with present plant design.		

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TABLE 4.6-1 (Cont'd)

Category	Component Parts to be Examined	Examination Method	Frequency of Examination	Examinations[1]
L-2	Pump casings	Visual	One pump of each type during 10-year interval if disassembled	One recirculating pump in 10 years.
M-1	Welds in valve bodies 3 inches and above	Visual and volumetric	One valve of each type during 10-year interval	Not applicable with present plant design
M-2	Valve bodies 3 inches and above	Visual	One valve of each type during 10-year interval if disassembled	One disassembled valve (with or without welds and 3 inches over in normal size) in each category and type shall be subject to visual examination. Individual examination shall cover 100% of the pressure boundary welds and may be performed at or near the end of the 10-year interval.
N	Interior surfaces and internals and integrally welded internal supports of the reactor vessel, including core spray spargers, core spray nozzles, and upper portions of jet pumps	Visual (not Inservice Inspection Code)	During first refueling outage and during subsequent refueling ages at approximately 3-year intervals	Interior surfaces and internal components of the reactor vessel, including the space at the bottom head and internal attachments which are welded to the vessel made accessible by the removal of components during normal refueling operations. All internal attachments whose failure may adversely affect core integrity shall be examined.
O	Control rod drive housing pressure-retaining welds.	Volumetric	The examinations performed during each inspection interval shall include the welds in 10% of the peripheral control rod drive housings.	The areas shall include the weld metal and base metal for one well thickness beyond the edge of the weld.

Notes

1. **Extent of Examinations**

Examinations which reveal unacceptable structural defects in a category shall be extended to include an additional number (or areas) of system components or piping in the same category approximately equal to those initially examined. In the event further unacceptable structural defects are revealed, all remaining system components or piping in the category shall be examined to the extent specified in that examination category.

(TSUP 4.0.E)

TABLE 4.6-1 (Cont'd)

2. Category J Weld Breakdown

Main Steamline - Group I Welds

Line	Weld Identification Unit 2
3001A-20-in.	30A-S10
3001B-20-in.	30B-S10
3001C-20-in.	30C-S10
3001D-20-in.	30D-S10

Group II Welds

Line	Weld Identification Unit 2
3001A-20-in.	30A-S20
	30A-F21
	30A-F24
3001B-20-in.	30B-S24
	30B-F25
	30B-F28
3001C-20-in.	30C-S21A
	30C-F22
	30C-F25
3001D-20-in.	30D-S21
	30D-F22
	30D-F25

Feedwater Line Group I Welds

Line	Weld Identification Unit 2
3204A-18-in.	32A-S4
3204B-18-in.	32B-S5

Group II Welds

Line	Weld Identification Unit 2
3204A-18-in.	32A-S1
	32A-F6
	32A-S1
3204B-18-in.	32B-F4
	32B-F7
3204C-12-in.	32C-S2
3204D-12-in.	32D-S2
	32D-S6
3204E-12-in.	32E-F7
	32D-S2
3204F-12-in.	32F-S2
	32F-F6

3. Supplemental Inspection Program for First and Second Refueling Outages

a. The following critical and sensitized components shall be nondestructively examined by the methods indicated:

Component	Examination Method
Bimetallic welds of field-replaced safe-ends	PT and (UT or RT)

b. The areas subject to examination shall include 100% of the exterior surfaces of the welds in Item 1. Weld areas to be examined shall include the base material for at least one wall thickness beyond the edge of the weld.

c. All examinations shall be conducted in accord with the examination techniques and procedures and meet the acceptance standards specified in the ASME Section XI Inservice Inspection Code and supplemented where necessary by special techniques with demonstrated capability to detect stress-corrosion cracking.

d. The examination frequency shall conform to the following schedule:

Bimetallic welds of field-replaced safe-ends

1) 25% at or within the first refueling outage 2) 25% at or within the second refueling outage

e. In the event any of the examinations for Item 4 reveal indications of structural defects which upon evaluation require repairs or replacements, the specified examination frequency shall be subject to review by the NRC.

TSUP 4.0.E

TABLE 4.6-2

REVISED WITHDRAWAL SCHEDULE FOR QUAD-CITIES UNIT 2

Withdrawal Year	Part No.	Location	Comments
1981	18	Wall - 215°	
2002	17	Wall - 95°	
	19	Wall - 245°	Standby
	15	Wall - 65°	Standby
	20	Wall - 275°	Standby
1979	14	Near Core Top Guide - 90°	
1981	16	Near Core Top Guide - 180°	

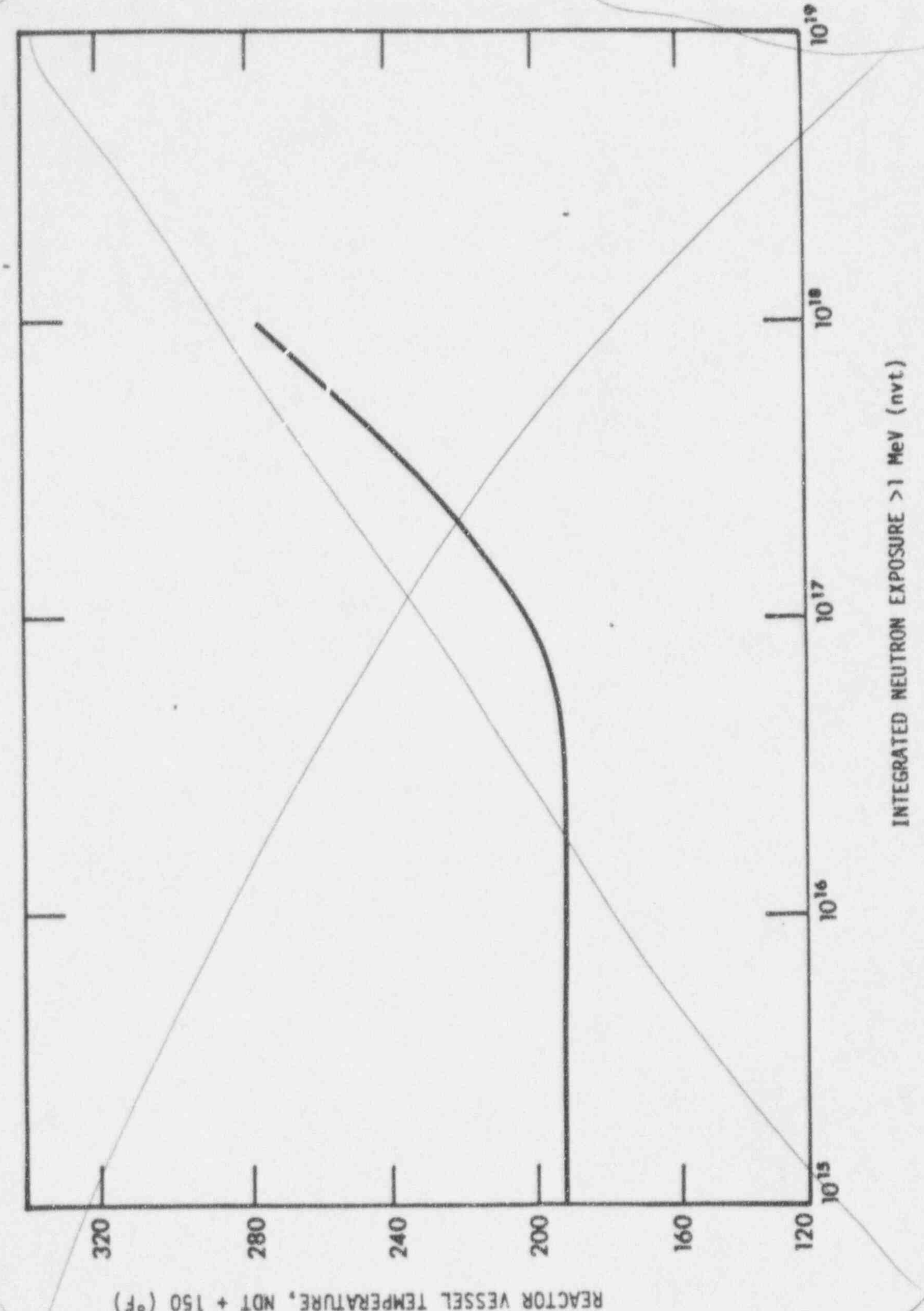


FIGURE 3.6-1

MINIMUM REACTOR
PRESSURIZATION TEMPERATURE

QUAD-CITIES
DPR-30

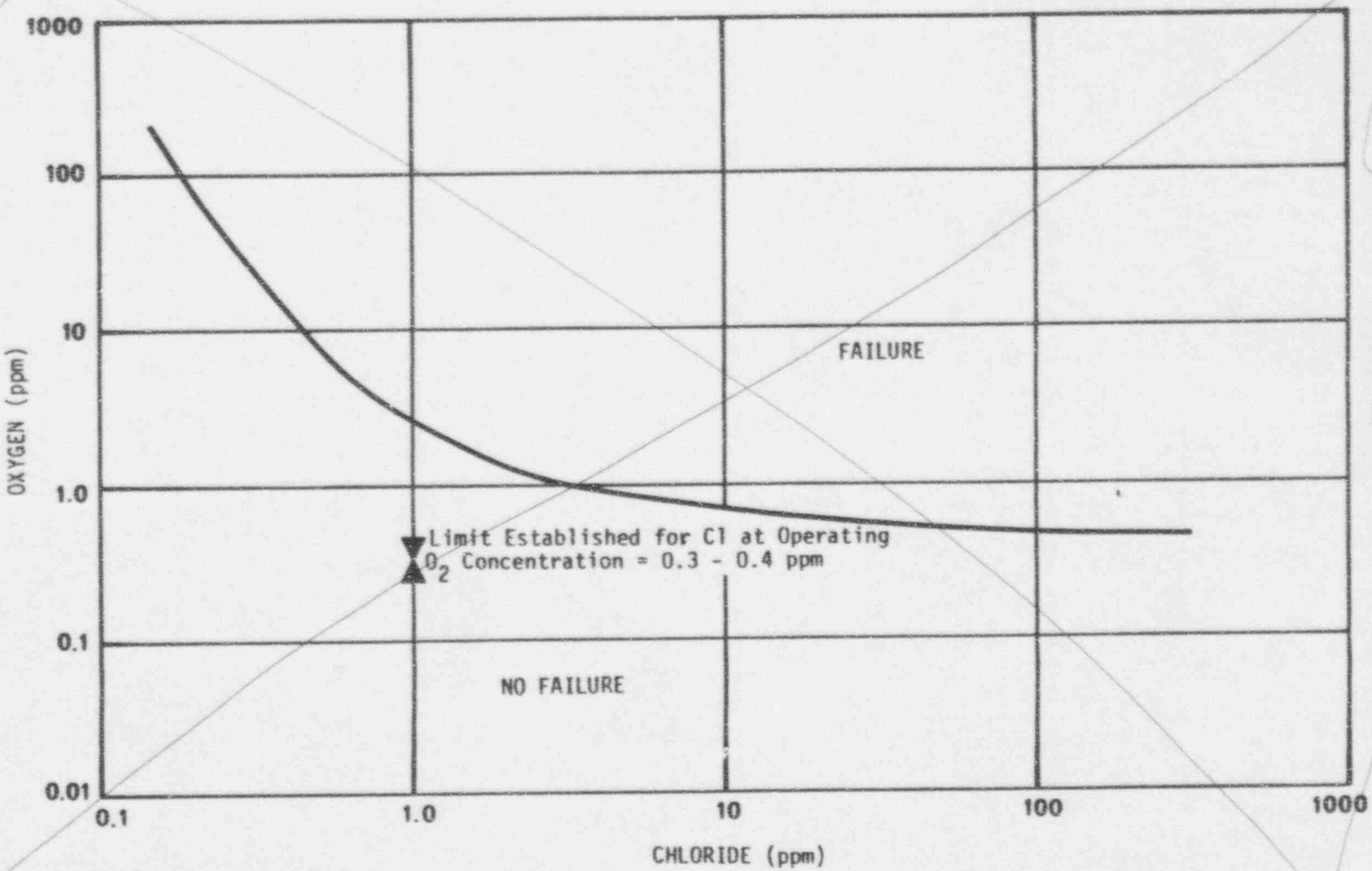


FIGURE 4.6-1
CHLORIDE STRESS CORROSION TEST
RESULTS AT 500 °F

ATTACHMENT E

Marked-Up BWR/4 STS Pages

FOR INFORMATION ONLY

3/4.4 REACTOR COOLANT SYSTEM

(6.A) 3/4.~~4~~.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

(6.A)

3.~~4~~.1.1 Two reactor coolant system recirculation loops shall be in operation.

APPLICABILITY: OPERATIONAL ~~CONDITIONS~~ MODE(S) 1* and 2*.

ACTION:

1 ~~a~~. With only one reactor coolant system recirculation loop not in operation, immediately initiate measures to place the unit in at least HOT SHUTDOWN within the next 12 hours.

2 ~~b~~. With no reactor coolant system recirculation loops in operation, immediately initiate measures to place the unit in at least STARTUP within 6 hours and in HOT SHUTDOWN within the next 6 hours.

(8)

(c. With a pump discharge bypass valve inoperable, verify the valve to be closed at least once per 31 days.)

within 24 hours either restore both loops to operation or

a. Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit by 0.01 per Specification 2.1.B, and

b. Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Operating Limit by 0.01 per Specification 3.11.c, and

c. Reduce the Average Power Range Monitor (APRM) Flow Biased Neutron Flux Scram and Rod Block and Rod Block Monitor Trip Setpoints to those applicable to single loop operation per Specifications 2.2.A and 3.2.E.

d. Reduce the AVERAGE LINEAR HEAT GENERATION RATE (APLHGR) to single loop operation limits as specified in the CORE OPERATING LIMITS REPORT.

e. Electrically prohibit the idle recirc pump from starting[a]

SURVEILLANCE REQUIREMENTS

4.~~4~~.1.1.1 Each pump discharge valve (and bypass valve) shall be demonstrated OPERABLE by cycling each valve through at least one complete cycle of full travel during each startup** prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER.

(6.A)

4.~~4~~.1.1.2 Each pump MG set scoop tube mechanical and electrical stop shall be demonstrated OPERABLE with the overspeed setpoints less than or equal to (112)% and (120)%, respectively, of rated core flow, at least once per 18 months.

specified in the CORE OPERATING LIMITS REPORT

*See Special Test Exception 3.10.4.
**If not performed within the previous 31 days.

(a) Except to permit testing in preparation for returning the pump to service.

INSERT FOR PAGE 3/4 4-1

FOR INFORMATION ONLY

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation with:

a. Total core flow greater than or equal to 45% of rated core flow, or

b. THERMAL POWER less than or equal to the limit specified in Figure 3.4.1.1-1.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

a. With one reactor coolant system recirculation loop not in operation, immediately initiate action to reduce THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1 within 2 hours and initiate measures to place the unit in at least HOT SHUTDOWN within 12 hours.

b. With no reactor coolant system recirculation loops in operation, immediately initiate action to reduce THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1 within 2 hours and initiate measures to place the unit in at least STARTUP within 6 hours and in HOT SHUTDOWN within the next 6 hours.

c. With two reactor coolant system recirculation loops in operation and total core flow less than 45% of rated core flow and THERMAL POWER greater than the limit specified in Figure 3.4.1.1-1:

1. Determine the APRM and LPRM** noise levels (Surveillance 4.4.1.1.3):

   a) At least once per 8 hours, and

   b) Within 30 minutes after the completion of a THERMAL POWER increase of at least 5% of RATED THERMAL POWER.

2. With the APRM or LPRM** neutron flux noise levels greater than three times their established baseline noise levels, immediately initiate corrective action to restore the noise levels to within the required limits within 2 hours by increasing core flow to greater than 45% of rated core flow or by reducing THERMAL POWER to less than or equal to the limit specified in Figure 3.4.1.1-1.

*See Special Test Exception 3.10.4.

**Detector levels A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of the core should be monitored.

FOR INFORMATION ONLY

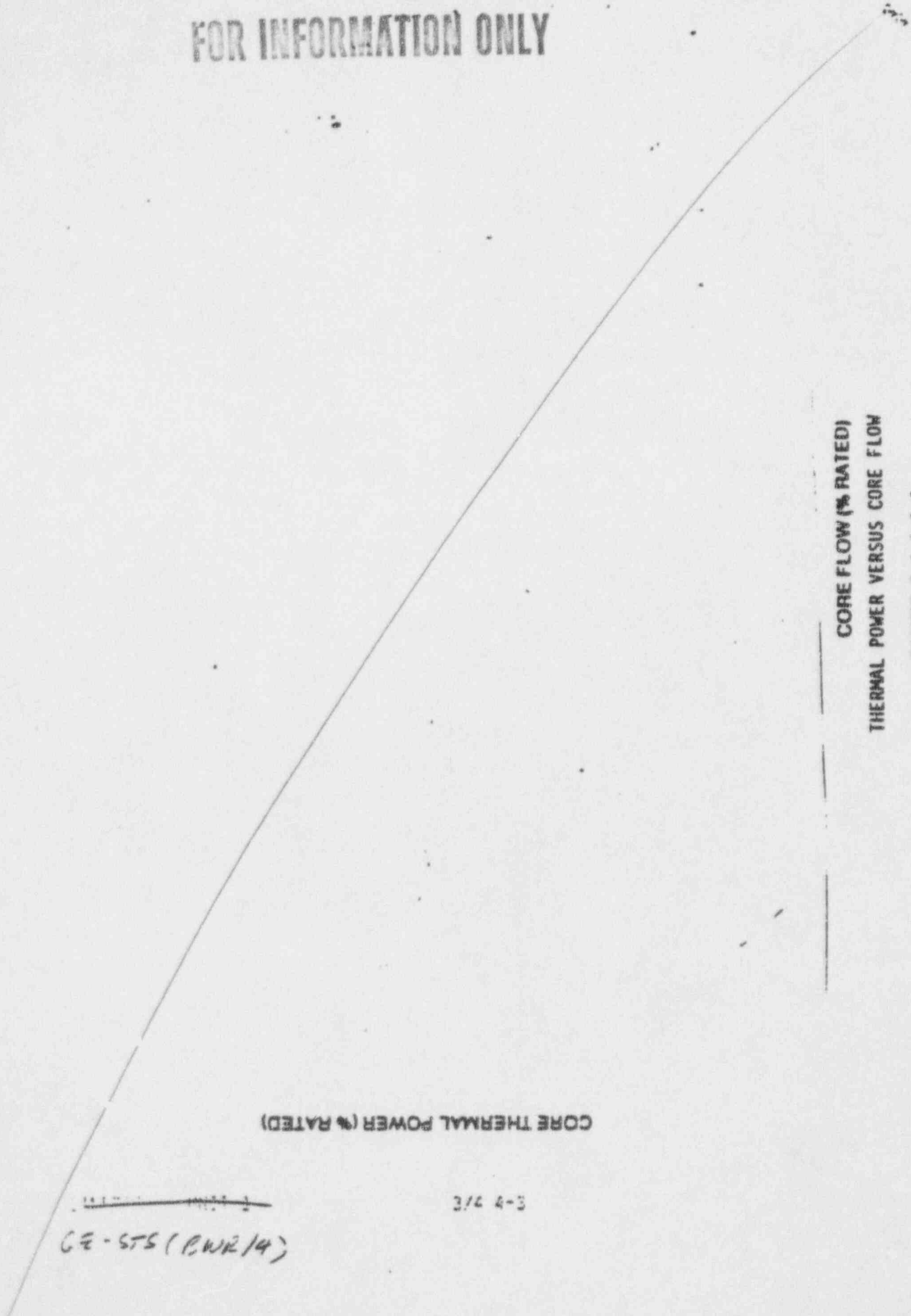
4.4.1.1.1 Each pump discharge valve shall be demonstrated OPERABLE by cycling each valve through at least one complete cycle of full travel during each startup* prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER.

4.4.1.1.2 Each pump MG set scoop tube mechanical and electrical stop shall be demonstrated OPERABLE with overspeed setpoints less than or equal to 105% and 102.5%, respectively, of rated core flow, at least once per 18 months.

4.4.1.1.3 Establish a baseline APRM and LPRM** neutron flux noise value within the regions for which monitoring is required (Specification 3.4.1.1, ACTION c) within 2 hours of entering the region for which monitoring is required unless baselining has previously been performed in the region since the last refueling outage.

*If not performed within the previous 31 days.

**Detector levels A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of the core should be monitored.



THERMAL POWER VERSUS CORE FLOW

FIGURE 3.4.1.1-1

GE-STS (BWR/4)

QUAD CITIES only

REACTOR COOLANT SYSTEM

JET PUMPS

FOR INFORMATION ONLY

LIMITING CONDITION FOR OPERATION

6.B

3.~~4~~e.1.2 All jet pumps shall be OPERABLE and flow indication shall be OPERABLE on at least 18 jet pumps[a].

APPLICABILITY: OPERATIONAL ~~CONDITIONS~~ MODE(S) 1 and 2.

ACTION:

1. With one or more jet pumps inoperable for other than inoperable flow indication, be in at least HOT SHUTDOWN within 12 hours.

2. With flow indication inoperable for three or more jet pumps, flow indication shall be restored such that at least 18 jet pumps have OPERABLE flow indication within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours.

3. With flow indication inoperable for both jet pumps in the same jet pump riser, flow indication shall be restored to OPERABLE status for at least one of these jet pumps within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours.

4. With flow indication inoperable on both calibrated (double tap) jet pumps in the same recirculation loop, flow indication shall be restored to OPERABLE status for at least one of these jet pumps within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.B

4.~~4~~.1.2 Each of the above required jet pumps shall be demonstrated OPERABLE prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER and at least once per 24 hours by determining recirculation loop flow, total core flow and diffuser-to-lower plenum differential pressure for each jet pump and verifying that no two of the following conditions occur when the recirculation pumps are operating at the same speed.

1. During two loop operation, at least once per 24 hours while greater than 25% of RATED THERMAL POWER, determine recirculation loop flow, total core flow and individual jet pump flow for each jet pump and verify that no two of the following occur when both recirculation pumps are operating in accordance with Specification 3.6.C

Similar to 4.6.B2

repeated for 4.6.B2

a. The indicated recirculation ~~loop~~ pump flow differs by more than 10% from the established ~~pump~~ speed-~~loop~~ flow ~~flow~~ characteristics.

b. The indicated total core flow differs by more than 10% from the established total core flow value derived from ~~recirculation loop flow measurements~~ established core plate ΔP/core flow relationships.

c. The indicated ~~diffuser-to-lower plenum differential pressure~~ flow of any individual jet pump differs from the established patterns by more than 10%.

d. The provisions of Specification 4.0.D are not applicable provided that the surveillance is performed within 24 hours after exceeding 25% of RATED THERMAL POWER.

X

FOR INFORMATION ONLY

RECIRCULATION PUMPS

LIMITING CONDITION FOR OPERATION

6.C

3.4.1.3 Recirculation pump speed shall be maintained within:

1. ~~a~~. 5% (10) of each other with core flow (THERMAL POWER) greater than or equal to 70% (80) of rated core flow.

2. ~~b~~. 10% (15) of each other with core flow less than 70% (80) of rated core flow (RATED THERMAL POWER)

APPLICABILITY: OPERATIONAL ~~CONDITIONS~~ MODE(S) 1* and 2*. during two recirculation loop operation

ACTION:

With the recirculation pump speeds different by more than the specified limits, either:

1. ~~a~~. Restore the recirculation pump speeds to within the specified limit within 2 hours, or

2. ~~b~~. Declare the recirculation loop of the pump with the slower speed not in operation (Trip one of the recirculation pumps) and take the ACTION required by Specification 3.4.1.1. (3.6.A.1)

SURVEILLANCE REQUIREMENTS

6.C

4.4.1.3 Recirculation pump speed shall be verified to be within the limits at least once per 24 hours.

*See Special Test Exception 3.10.4.

IDLE RECIRCULATION LOOP STARTUP

LIMITING CONDITION FOR OPERATION

6.D

3.~~4~~.1.4 An idle recirculation loop shall not be started unless the temperature differential between the reactor pressure vessel steam space coolant and the bottom head drain line coolant is less than or equal to (~~100~~)°F, and: 145°F[a]

1. ~~a~~. When both loops have been idle, unless the temperature differential between the reactor coolant within the idle loop to be started up and the coolant in the reactor pressure vessel is less than or equal to (50)°F, or 50

2. ~~b~~. When only one loop has been idle, unless the temperature differential between the reactor coolant within the idle and operating recirculation loops is less than or equal to (50)°F and the ~~operating loop flow rate is less than or equal to (50)% of rated loop flow.~~ speed of the operating pump is ≤ 43% of rated pump speed.

APPLICABILITY: OPERATIONAL ~~CONDITIONS~~ MODE(S) 1, 2, 3 and 4.

ACTION:

With temperature differences and/or flow rates exceeding the above limits, suspend startup of any idle recirculation loop.

SURVEILLANCE REQUIREMENTS

6.D

4.~~4~~.1.4 The temperature differentials and flow rate shall be determined to be within the limits within 15 minutes prior to startup of an idle recirculation loop.

[a] Below 25 psig reactor pressure, this temperature differential is not applicable.

FOR INFORMATION ONLY

6.E

3/4.4.2 SAFETY/RELIEF VALVES

SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

6.E

3.4.2.1 (At least (two) reactor coolant system code safety valves and) the safety valve function of at least (11) (of the following) reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings:* (a) in accordance the 9

established as

(2) safety valves @ (1146) psig ±1% (b)
1 (3) safety-relief valves @ (1175) psig ±1% 1135
2 (3) safety-relief valves @ (1185) psig ±1% 1240
2 (3) safety-relief valves @ (1195) psig ±1% 1250
4 (2) safety-relief valves @ (1205) psig ±1% 1260

APPLICABILITY: OPERATIONAL ~~CONDITIONS~~ MODE(S) 1, 2 and 3.

Each installed safety valve shall be closed with OPERABLE position indication

ACTION:

1. ~~a.~~ With (one or more of the above required reactor coolant system code safety valves or with) the safety valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more (code safety valves or) safety/relief valves stuck open, provided that suppression pool average water temperature is less than (95)°F, close the stuck open (code safety valves and/or) safety relief valve(s); if unable to close the stuck open valve(s) within 2 minutes or if suppression pool average water temperature is (95)°F or greater, place the reactor mode switch in the Shutdown position.

c. With one or more safety/relief valve (tail-pipe pressure switches) (acoustic monitors) inoperable, restore the inoperable (switch(es)) (monitor(s)) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With all position indication inoperable on one or more safety valve(s), restore the inoperable position indication to OPERABLE status within 30 days or be in HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours

(a) *The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

(b) Target Rock combination safety/relief valves

SURVEILLANCE REQUIREMENTS

(4.4.2.1.1 (The code safety valve function of each of the above required safety/relief valves shall be demonstrated OPERABLE by verifying that the bellows on the safety/relief valves have integrity, by instrumentation indication, at least once per 24 hours.)

6.E.1

position indicators

4.4.2.1.2 The (tail-pipe pressure switch) (acoustic monitor) for each safety/relief valve shall be demonstrated OPERABLE with the setpoint verified to be ((20) ± (5) psig) by performance of a:

a. CHANNEL (FUNCTIONAL TEST) (CHECK) at least once per 31 days, and a

b. CHANNEL CALIBRATION at least once per 18 months(*).

(*The provisions of Specification 4.0.4 are not applicable provided the Surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.)

6.E.2

once per 12 months

4.4.2.2 At least 1/2 of the safety relief valves shall be removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations, at least once per 18 months, and they shall be rotated such that all 14 safety relief valves are removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations tested at least once per 40 months.

9

At least once per 40 months, the safety valves

FOR INFORMATION ONLY

REACTOR COOLANT SYSTEM

SAFETY/RELIEF VALVES LOW-LOW SET FUNCTION

5 reactor coolant system relief valves and the reactuation time delay of two

LIMITING CONDITION FOR OPERATION

(G.F)

3.4.2.2 The relief valve function and the low-low set function of the following reactor coolant system safety/relief valves shall be OPERABLE with the following settings:

Each installed relief valve shall be closed with OPERABLE position indication

Valve No.	Low-Low Set Function Setpoint* (psig) ± 1% Open	Close	Relief Function Setpoint* (psig) ± 1% Open	Close
______	(1033)	(926)	≤ 1115 psig	______
______	(1073)	(936)	≤ 1115 psig	______
______	(1113)	(946)	≤ 1135 psig	______
______	(1113)	(946)	≤ 1135 psig	______
______	(1113)	(946)	≤ 1135 psig (a)	______

APPLICABILITY: OPERATIONAL ~~CONDITIONS~~ MODE(S) 1, 2 and 3.

ACTION:

1. With one or more relief valves open, provided that suppression pool average water temperature is < 110°F, take action to close the open relief valve(s). If suppression pool average water temperature is ≥ 110°F place the reactor mode switch in the Shutdown position.

2. ~~a.~~ With the relief valve function and/or the low-low set function reactuation time delay of one of the above required reactor coolant system safety/relief valves inoperable, restore the inoperable relief valve function and low-low set function to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. ~~b.~~ With the relief valve function and/or the low-low set function reactuation time delay of more than one of the above required reactor coolant system safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

4. With all position indication inoperable on one or more relief valve(s), restore the inoperable position indication to OPERABLE status within 30 days or be in HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

reactuation time delay function

(G.F)

4.4.2.2.1 The relief valve function and the low-low set function pressure actuation instrumentation shall be demonstrated OPERABLE by performance of a:

of the relief valve function

a. CHANNEL FUNCTIONAL TEST, including calibration of the trip unit, at least once per 31 (92) days.

b. CHANNEL CALIBRATION, LOGIC SYSTEM FUNCTIONAL TEST and simulated automatic operation of the entire system at least once per 18 months.

(a) Target Rock combination safety/relief valve

*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

2. A position indicator for each relief valve shall be demonstrated OPERABLE by performance of a:

a. CHANNEL CHECK at least once per 31 days, and a

b. CHANNEL CALIBRATION at least once per 18 months.

REACTOR COOLANT SYSTEM **FOR INFORMATION ONLY**

4.5.2 e

3/4.~~4.3~~ REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.3.1 The following reactor coolant system leakage detection systems shall be OPERABLE:

1. ~~a.~~ The primary containment atmosphere (gaseous or particulate) radioactivity monitoring system, e

2. ~~b.~~ The ~~primary containment~~ drywell floor drain sump flow monitoring sampling system, and e

c. Either the (primary containment air coolers condensate flow rate monitoring system) or the primary containment atmosphere (gaseous or particulate) radioactivity monitoring system.

APPLICABILITY: OPERATIONAL ~~CONDITIONS~~ MODE(s) 1, 2 and 3.

ACTION:

a. With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous and/or particulate radioactive monitoring system is inoperable; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. e

1. With the primary containment atmosphere [illegible] system inoperable, restore the inoperable leak detection radioactivity sampling system to OPERABLE status within 24 hours; otherwise be in HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
2. With the drywell floor drain sump system inoperable, restore the drywell floor drain system to OPERABLE status within 24 hours; otherwise be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.5.2 e

4.~~4.3.1~~ The reactor coolant system leakage detection systems shall be demonstrated OPERABLE by:

1. ~~a.~~ Primary containment atmosphere particulate and gaseous monitoring systems-performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months. e

b. Primary containment sump flow monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION TEST at least once per 18 months.

c. Primary containment air coolers condensate flow rate monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

1. Performing the leakage determination of Specification 4.6.H.
2. Performing a CHANNEL CALIBRATION of the drywell floor drain sump pump discharge flow integrator at least once per 18 months.

 FOR INFORMATION ONLY

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

(6.H)

3.~~4.3.2~~ Reactor coolant system leakage shall be limited to:

1. ~~a.~~ No PRESSURE BOUNDARY LEAKAGE.

3. ~~b.~~ 5 gpm UNIDENTIFIED LEAKAGE.

2. ~~c.~~ ≤25 gpm total leakage averaged over any 24-hour surveillance period.

d. 1 gpm leakage at a reactor coolant system pressure of (950) ±(10) psig from any reactor coolant system pressure isolation valve specified ~~in Table 3.4.3.2-1.~~

4. (e.) ≤2 gpm increase in UNIDENTIFIED LEAKAGE within any ~~4-hour~~ period of 24 hours or less (Applicable in OPERATIONAL MODE(s) 1 only).

APPLICABILITY: OPERATIONAL MODE(S) ~~CONDITIONS~~ 1, 2 and 3.

ACTION:

1. ~~a.~~ With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

2. ~~b.~~ With the ~~any~~ reactor coolant system UNIDENTIFIED LEAKAGE or total leakage rate(s) ~~leakage~~ greater than the above limits ~~in b and/or c, above~~, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With any reactor coolant system pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two other closed (manual or deactivated automatic) (or check*) valves, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With one or more of the high/low pressure interface valve leakage pressure monitors shown in Table 3.4.3.2-1 inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm setpoint at least once per 12 hours; restore the inoperable monitor(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. (e.) With an increase in reactor coolant system UNIDENTIFIED LEAKAGE ~~any reactor coolant system UNIDENTIFIED LEAKAGE increase~~ greater than 2 gpm within any period of 24 hours or less in OPERATIONAL MODE 1: ~~4-hour period~~, identify the source of leakage increase as not IGSCC susceptible material ~~service sensitive Type 304 or 316 austenitic stainless steel~~ within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.)

(*Which have been verified not to exceed the allowable leakage limit at the last refueling outage or the after last time the valve was disturbed, whichever is more recent.)

FOR INFORMATION ONLY

SURVEILLANCE REQUIREMENTS

6.H

4.4.3.2.1 The reactor coolant system leakage shall be demonstrated to be within each of the ~~above~~ limits by:

1. ~~a.~~ Sampling ~~Monitoring~~ the primary containment atmospheric (particulate) (and) (gaseous) radioactivity at least once per (4) (12) hours, (a)

2. ~~b.~~ Determining ~~Monitoring~~ the primary containment sump flow rate at least once per (4) (12) hours. (and) 8 hours, not to exceed 12 hours.

c. Monitoring the primary containment air coolers condensate flow rate or the (gaseous) (particulate) radioactivity at least once per (4) (12) hours, and

d. Monitoring the reactor vessel head flange leak detection system at least once per 24 hours.

4.4.3.2.2 Each reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1 shall be demonstrated OPERABLE by leak testing pursuant to Specification 4.0.5 and verifying the leakage of each valve to be within the specified limit:

a. At least once per 18 months, and

b. Prior to returning the valve to service following maintenance, repair or replacement work on the valve which could affect its leakage rate.

The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

4.4.3.2.3 The high/low pressure interface valve leakage pressure monitors shall be demonstrated OPERABLE with alarm setpoints per Table 3.4.3.2-2 by performance of a:

a. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

b. CHANNEL CALIBRATION at least once per 18 months.

(a) Not a means of quantifying leakage

[illegible] FOR INFORMATION ONLY

TABLE 3.4.3.2-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	SYSTEM

TABLE 3.4.3.2-2

REACTOR COOLANT SYSTEM INTERFACE VALVES

LEAKAGE PRESSURE MONITORS

VALVE NUMBER	SYSTEM	ALARM SETPOINT (psig)

 FOR INFORMATION ONLY

(6.1) 3/4.~~4.4~~ CHEMISTRY

LIMITING CONDITION FOR OPERATION

(6.1) 3.~~4.4~~ The chemistry of the reactor coolant system shall be maintained within the limits specified in Table ~~3.4.4-1~~. (3.6.I-1)

APPLICABILITY: ~~At all times.~~ 1, 2(a), and 3(a)

ACTION:

1. ~~a.~~ In OPERATIONAL ~~CONDITION~~ MODE 1:

   a. ~~1.~~ With the conductivity, chloride concentration or pH exceeding the limit specified in Table ~~3.4.4-1~~ (3.6.I-1) for less than 72 hours during one continuous time interval and, for conductivity and chloride concentration, for less than 336 hours per year, but with the conductivity less than 10 μmho/cm at 25°C and with the chloride concentration less than 0.5 ppm, this need not be reported to the Commission and the provisions of Specification 3.0.4 are not applicable.

   b. ~~2.~~ With the conductivity, chloride concentration or pH exceeding the limit specified in Table ~~3.4.4-1~~ for more than 72 hours during one continuous time interval or with the conductivity and chloride concentration exceeding the limit specified in Table ~~3.4.4-1~~ (3.6.I-1) for more than 336 hours per year, be in at least STARTUP within the next (8) ~~6~~ hours.

   c. ~~3.~~ With the conductivity exceeding 10 μmho/cm at 25°C or chloride concentration exceeding 0.5 ppm, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

2. ~~b.~~ In OPERATIONAL ~~CONDITION~~ MODE(S) 2 and 3 with the conductivity, chloride concentration or pH exceeding the limit specified in Table ~~3.4.4-1~~ (3.6.I-1) for more than 48 hours during one continuous time interval, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. ~~c.~~ At all other times:

   a. ~~1.~~ With the:

      1. ~~a)~~ Conductivity or pH exceeding the limit specified in Table ~~3.4.4-1~~ (3.6.I-1), restore the conductivity and pH to within the limit within 72 hours, or

      2. ~~b)~~ Chloride concentration exceeding the limit specified in Table ~~3.4.4-1~~ (3.6.I-1), restore the chloride concentration to within the limit within 24 hours, or

      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 structural integrity of the reactor coolant system remains acceptable for continued operation prior to proceeding to OPERATIONAL ~~CONDITION~~ MODE 3.

   b. ~~2.~~ The provisions of Specification 3.0.3 are not applicable.

(a) the provisions of Specification 3.0.3 are not applicable during unit shutdown when entering OPERATIONAL MODE(S) 2 and 3 from OPERATIONAL MODE 1

 FOR INFORMATION ONLY

SURVEILLANCE REQUIREMENTS

(6.I)
4.~~4.4~~ The reactor coolant shall be determined to be within the specified chemistry limit by:

1. ~~a~~. Measurement prior to pressurizing the reactor during each startup, if not performed within the previous 72 hours.

2. ~~b~~. Analyzing a sample of the reactor coolant for:

  a ~~1~~. Chlorides at least once per:

    1 ~~a)~~ 72 hours, and

    2 ~~b)~~ 8 hours whenever conductivity is greater than the limit in Table ~~3.4.4-1~~. 3.6.I-1

  b ~~2~~. Conductivity at least once per 72 hours.

  c ~~3~~. pH at least once per:

    a) 72 hours, and

    b) 8 hours whenever conductivity is greater than the limit in Table ~~3.4.4-1~~. 3.6.I-1

3. ~~c~~. Continuously recording the conductivity of the reactor coolant, or, when the continuous recording conductivity monitor is inoperable for up to 31 days, obtaining an in-line conductivity measurement at least once per:

  a ~~1~~. 4 hours in OPERATIONAL ~~CONDITIONS~~ MODE(s) 1, 2 and 3, and

  b ~~2~~. 24 hours at all other times.

4. ~~d~~. Performance of a CHANNEL CHECK of the continuous conductivity monitor with an in-line flow cell at least once per:

  a ~~1~~. 7 days, and

  b ~~2~~. 24 hours whenever conductivity is greater than the limit in in Table ~~3.4.4-1~~. 3.6.I-1

6.I-1

TABLE 3.4.4-1

REACTOR COOLANT SYSTEM
CHEMISTRY LIMITS

MODE(S) ~~OPERATIONAL CONDITION~~	CHLORIDES	CONDUCTIVITY (μmhos/cm @25°C)	PH
1	≤ 0.2 ppm	≤ 1.0	5.6 ≤ pH ≤ 8.6
2 and 3	≤ 0.1 ppm	≤ 2.0	5.6 ≤ pH ≤ 8.6
At all other times	≤ 0.5 ppm	≤ 10.0	5.3 ≤ pH ≤ 8.6

3/4.4.5 SPECIFIC ACTIVITY

6.J

LIMITING CONDITION FOR OPERATION

3.4.5 The specific activity of the primary coolant shall be limited to:

reactor

a. Less than or equal to 0.2 microcuries per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to 100/$\bar{E}$ microcuries per gram.

APPLICABILITY: OPERATIONAL ~~CONDITIONS~~ MODE(S) 1, 2, 3 and 4.

ACTION:

1. a. In OPERATIONAL ~~CONDITIONS~~ MODE(S) 1, 2 or 3 with the specific activity of the primary coolant: (reactor)

1. Greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 but less than or equal to 4.0 microcuries per gram, operation may continue for up to 48 hours provided that the cumulative operating time under these circumstances does not exceed 800 hours in any consecutive 12-month period. With the total cumulative operating time at a primary coolant specific activity greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive six-month period, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days indicating the number of hours of operation above this limit. The provisions of Specification 3.0.4 are not applicable.

2. Greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or for more than 800 hours cumulative operating time in a consecutive 12-month period, or greater than 4.0 microcuries per gram; be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours. (DOSE EQUIVALENT I-131)

3. Greater than 100/$\bar{E}$ microcuries per gram, be in at least HOT SHUTDOWN with the main steamline isolation valves closed within 12 hours.

Special Reporting

b. In OPERATIONAL CONDITIONS 1, 2, 3 or 4, with the specific activity of the primary coolant greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 or greater than 100/$\bar{E}$ microcuries per gram, perform the sampling and analysis requirements of Item 4a of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit. A REPORTABLE OCCURRENCE shall be prepared and submitted to the Commission pursuant to Specification 6.9.1. This report shall contain the results of the specific activity analyses and the time duration when the specific activity of the coolant exceeded 0.2 microcuries per gram DOSE EQUIVALENT I-131 together with the following additional information.

(reactor) (6.J-1)

ACTION (Continued)

3 ~~c~~. In OPERATIONAL ~~CONDITION~~ MODE(S) 1 or 2, with:

a ~~1~~. THERMAL POWER changed by more than ~~15~~ 20% of RATED THERMAL POWER in one hour*, or

b ~~2~~. The off-gas level, at the SJAE, increased by more than ~~(10,000)~~ 25,000 microcuries per second in one hour during steady state operation at release rates less than ~~(75,000)~~ 100,000 microcuries per second, or

3. The off-gas level, at the SJAE, increased by more than (15)% in one hour during steady state operation at release rates greater than ~~(75,000)~~ 100,000 microcuries per second,

perform the sampling and analysis requirements of Item ~~4b~~ 3b of Table ~~4.4.5-1~~ 4.6.J-1 until the specific activity of the ~~primary~~ reactor coolant is restored to within its limit. Prepare and submit to the Commission a Special Report pursuant to Specification 6.9.2 at least once per 92 days containing the results of the specific activity analysis together with the below additional information for each occurrence.

Additional Information

1. Reactor power history starting 48 hours prior to:

   a) The first sample in which the limit was exceeded, and/or

   b) The THERMAL POWER or off-gas level change.

2. Fuel burnup by core region.

3. Clean-up flow history starting 48 hours prior to:

   a) The first sample in which the limit was exceeded, and/or

   b) The THERMAL POWER or off-gas level change.

4. Off-gas level starting 48 hours prior to:

   a) The first sample in which the limit was exceeded, and/or

   b) The THERMAL POWER or off-gas level change.

SURVEILLANCE REQUIREMENTS

~~4.4.5~~ 6.J The specific activity of the reactor coolant shall be demonstrated to be within the limits by performance of the sampling and analysis program of Table ~~4.4.5-1~~ 4.6.J-1.

*Not applicable during the startup test program.

GE-STS (BWR/4)

TABLE 4.4.5-1

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	OPERATIONAL ~~CONDITIONS~~ MODE(S) IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Beta and Gamma Activity Determination	At least once per 72 hours	1, 2, 3
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	At least once per 31 days	1
3. Radiochemical for Ē Determination	At least once per 6 months*	1
3. ~~4.~~ Isotopic Analysis for Iodine	a) At least once per 4 hours, whenever the specific activity exceeds a limit, as required by ACTION ~~b~~. 2	1#, 2#, 3#, 4# (2)
	b) At least one sample, between 2 and 6 hours following the change in THERMAL POWER or off-gas level, as required by ACTION ~~c~~. 3	1, 2
4. ~~5.~~ Isotopic Analysis of an Off-gas Sample Including Quantitative Measurements for at least Xe-133, Xe-135 and Kr-88	At least once per 31 days	1

*Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was ~~last subcritical for 48 hours or longer.~~

(2) #Until the specific activity of the ~~primary~~ reactor coolant system is restored to within its limits.

3/4 4-~~17~~ 19

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6.K

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

6.K 6.K-1

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

1. ~~a.~~ A maximum reactor coolant heatup of (100)°F in any one hour period,

2. ~~b.~~ A maximum reactor coolant cooldown of (100)°F in any one hour period,

3. ~~c.~~ A maximum reactor coolant 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, and

4. ~~d.~~ The reactor vessel flange and head flange temperature greater than or equal to ~~(70)~~ 100 °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

6.K.1

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 3.4.6.1-1 curves A and A', B and B', or C and C' as applicable, at least once per 30 minutes.

6.K-1 or B

SURVEILLANCE REQUIREMENTS (Continued)

(b. The reactor coolant system temperature at the following location shall be determined at least once per 5 minutes until 3 successive temperatures at each location are within 5°F:

1. Reactor vessel bottom drain,
2. Recirculation loops A and B, and
3. Reactor vessel bottom head.)

(6.K.3) 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 3.4.6.1-1 curves C and C' 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. (6.K-1)

(6.K.3) 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 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.

(6.K.4) 4.4.6.1.4 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to (70)°F: 100

a. In OPERATIONAL ~~CONDITION~~ MODE 4 when reactor coolant system temperature is:

   1. ≤ 100°F, at least once per 12 hours. 130

   2. ≤ (80)°F, at least once per 30 minutes. 110

b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.

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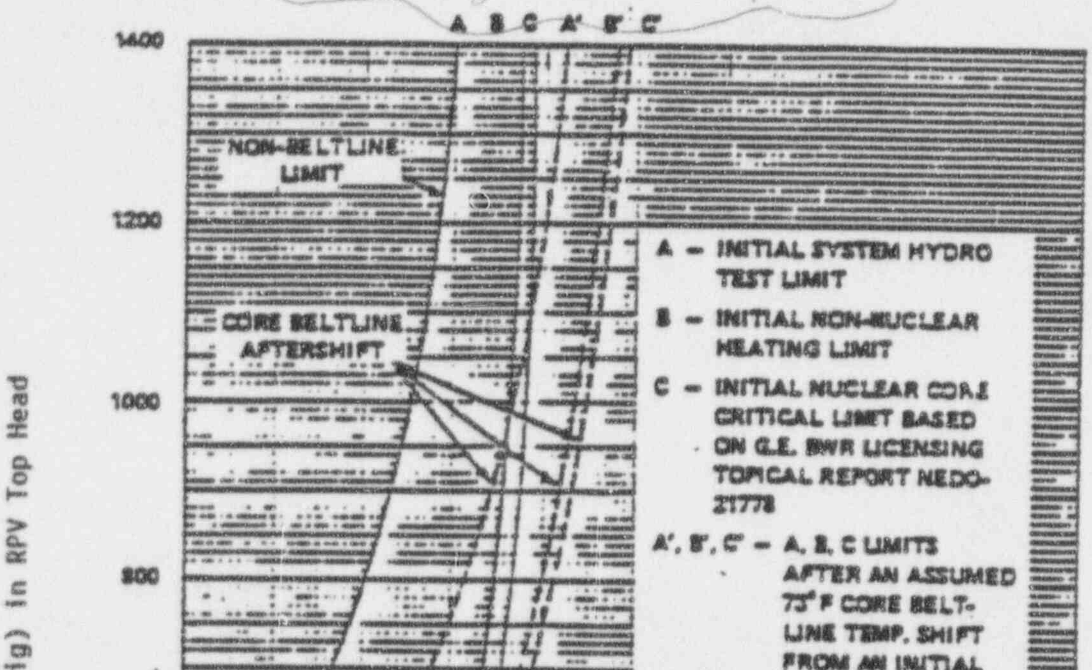
Dresden/Quad C - specific figures incorporated



MINIMUM (REACTOR PRESSURE VESSEL METAL) TEMPERATURE VS. REACTOR VESSEL PRESSURE

Figure 3.4.6.1-1

TABLE 4.4.6.1.3-1

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM-WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)

2 3

REACTOR COOLANT SYSTEM FOR INFORMATION ONLY

REACTOR STEAM DOME

LIMITING CONDITION FOR OPERATION

(6.2)

3.4.6.2 The pressure in the reactor steam dome shall be less than (1045) psig.

or equal to 1005

APPLICABILITY: OPERATIONAL CONDITION 1* and 2*.

ACTION:

1005

With the reactor steam dome pressure exceeding (1045) psig, reduce the pressure to less than (1045) psig within 15 minutes or be in at least HOT SHUTDOWN within 12 hours.

or equal to 1005

SURVEILLANCE REQUIREMENTS

(6.6)

4.4.6.2 The reactor steam dome pressure shall be verified to be less than (1045) psig at least once per 12 hours.

or equal to 1005

*Not applicable during anticipated transients.

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REACTOR COOLANT SYSTEM

6.M

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.4.7 Two main steam line isolation valves (MSIVs) per main steam line shall be OPERABLE with closing times greater than or equal to (3) and less than or equal to (5) seconds.

APPLICABILITY: OPERATIONAL ~~CONDITIONS~~ MODES 1, 2 and 3.

ACTION:

a. With one or more MSIVs inoperable:

1. Maintain at least one MSIV OPERABLE in each affected main steam line that is open and within 8 hours, either:

   a) Restore the inoperable valve(s) to OPERABLE status, or

   b) Isolate the affected main steam line by use of a deactivated MSIV in the closed position.

2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. T[illegible]ovisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

6.M

4.4.7 Each of the above required MSIVs shall be demonstrated OPERABLE by verifying full closure between (3) and (5) seconds when tested pursuant to Specification 4.0.5.

E

(6.N) 3/4.4.8 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

(6.N) 3.4.8 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.8. (6.N)

APPLICABILITY: OPERATIONAL MODE(S) ~~CONDITIONS~~ 1, 2, 3, 4 and 5.

ACTION:

1 ~~a~~. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

2 ~~b~~. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F. e

3 ~~c~~. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

4 ~~d~~. The provisions of Specification 3.0.4 are not applicable. e

SURVEILLANCE REQUIREMENTS

(6.N) 4.4.8 No requirements other than Specification 4.0.5. (E)

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ONM

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6.0 REACTOR COOLANT SYSTEM

3/4.4.9 RESIDUAL HEAT REMOVAL shutdown Cooling

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

6.0 3.4.9.1 Two# (a) shutdown cooling mode (SDC) loops of the residual heat removal (RHR) system shall be OPERABLE and, unless at least one recirculation pump is in operation, at least one shutdown cooling mode loop shall be in operation*,## (b) (c), with each loop consisting of at least:

1. ~~a.~~ One OPERABLE RHR SDC pump, and

2. ~~b.~~ One OPERABLE RHR SDC heat exchanger.

APPLICABILITY: OPERATIONAL ~~CONDITION~~ MODE 3, with reactor vessel pressure coolant temperature less than the RHR SDC cut-in permissive setpoint.

ACTION:

1. ~~a.~~ With less than the above required RHR shutdown cooling mode (SDC) loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible. Within one hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode (SDC) loop. Be in at least COLD SHUTDOWN within 24 hours.** (d)

2. ~~b.~~ With no RHR shutdown cooling mode loop in operation, immediately initiate corrective action to return at least one ~~loop~~ shutdown cooling loop or recirculation pump to operation as soon as possible. Within one hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour. RHR shutdown cooling loop or recirculation pump

SURVEILLANCE REQUIREMENTS

6.0 4.4.9.1 At least one shutdown cooling mode (SDC) loop of the residual heat removal system or alternate method shall be determined to be in operation and circulating reactor coolant at least once per 12 hours. one recirculation pump,

(a) #One RHR shutdown cooling mode loop may be inoperable for up to 2 hours for surveillance testing provided the other loop is OPERABLE and in operation.

(b) A *The shutdown cooling pump may be removed from operation for up to 2 hours per 8 hour period provided the other loop is OPERABLE.

(c) ##The RHR shutdown cooling mode loop may be removed from operation during hydrostatic testing.

(d) **Whenever two or more RHR SDC subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

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REACTOR COOLANT SYSTEM

COLD SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.9.2 Two# shutdown cooling (SDC) ~~mode~~ loops of the residual heat removal (RHR) system shall be OPERABLE and, unless at least one recirculation pump is in operation, at least one shutdown cooling ~~mode~~ loop shall be in operation*,## with each loop consisting of at least:

1. ~~a.~~ One OPERABLE ~~RHR~~ SDC pump, and
2. ~~b.~~ One OPERABLE ~~RHR~~ heat exchanger.

APPLICABILITY: OPERATIONAL ~~CONDITION~~ MODE 4.

ACTION:

1. ~~a.~~ With less than the above required SDC ~~RHR shutdown cooling mode~~ loops OPERABLE, within one hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable ~~RHR shutdown cooling mode~~ loop.
2. ~~b.~~ With no ~~RHR~~ shutdown (SDC) cooling ~~mode~~ loop or recirculation pump in operation, within one hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

SURVEILLANCE REQUIREMENTS

4.4.9.2 At least one SDC ~~shutdown cooling mode~~ loop ~~of the residual heat removal system~~, recirculation pump or alternate method shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

(a) ~~#~~One ~~RHR~~ shutdown cooling ~~mode~~ loop may be inoperable for up to 2 hours for surveillance testing provided the other loop is OPERABLE and in operation.

(b) ~~*~~The shutdown cooling pump may be removed from operation for up to 2 hours per 8 hour period provided the other loop is OPERABLE.

(c) ~~##~~The shutdown cooling ~~mode~~ loop may be removed from operation during hydrostatic testing.

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3/4 4.9 RESIDUAL HEAT REMOVAL

6.0

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.9.1 Two# shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE (a) and, unless at least one recirculation pump is in operation, at least one shutdown cooling mode loop subsystem shall be in operation*,## capable of circulating reactor coolant (b), with each loop consisting of at least:

a. One OPERABLE RHR pump, and

b. One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL ~~CONDITION~~ MODE 3, with reactor vessel pressure less than the RHR cut-in permissive setpoint.

ACTION:

1 ~~a~~. With less than the above required RHR shutdown cooling mode loops subsystems OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible. Within one hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode loop subsystem. Be in at least COLD SHUTDOWN within 24 hours.** (c)

2 ~~b~~. With no RHR shutdown cooling mode loop OPERABLE in operation, immediately initiate corrective action to return at least one ~~loop~~ subsystem to operation OPERABLE status as soon as possible. Within one hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour. RHR shutdown cooling ... loop or recirculation pump

SURVEILLANCE REQUIREMENTS

(6.0) 4.4.9.1 At least one shutdown cooling mode loop subsystem of the residual heat removal system or recirculation pump, or alternate method shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

#One RHR shutdown cooling mode loop may be inoperable for up to 2 hours for surveillance testing provided the other loop is OPERABLE and in operation.

*The shutdown cooling pump may be removed from operation for up to 2 hours per 8 hour period provided the other loop is OPERABLE.

(b) ##The RHR shutdown cooling ~~mode loop~~ pump may be removed from operation during hydrostatic testing.

(c) **Whenever two or more RHR SPC mode subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

(a) Each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. The provisions of Specification 3.0.D are not applicable.

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REACTOR COOLANT SYSTEM

COLD SHUTDOWN

LIMITING CONDITION FOR OPERATION

(c.P) (a) (subsystems)

3.4.9.2 Two# shutdown cooling mode ~~loops~~ subsystems of the residual heat removal (RHR) system shall be OPERABLE and, unless at least one recirculation pump is in operation, at least one shutdown cooling mode ~~loop~~ shall be ~~in operation~~*,## with each ~~loop~~ subsystem consisting of at least:

(capable of circulating reactor coolant (a))

1. ~~a.~~ One OPERABLE RHR pump, and

2. ~~b.~~ One OPERABLE RHR heat exchanger.

APPLICABILITY: OPERATIONAL ~~CONDITION~~ MODE 4.

ACTION:

(subsystems)

1. ~~a.~~ With less than the above required RHR shutdown cooling mode ~~loops~~ OPERABLE, within one hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode ~~loop~~. (subsystem)

(subsystem OPERABLE, immediately initiate corrective action to return at least one subsystem to OPERABLE status as soon as possible.)

(with a recirculation pump)

~~b.~~ With no RHR shutdown cooling mode ~~loop~~ in operation, within one hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

SURVEILLANCE REQUIREMENTS

(c.P) (subsystem)

4.4.9.2 At least one shutdown cooling mode ~~loop~~ of the residual heat removal system or alternate method shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

(recirculation pump)

#One RHR shutdown cooling mode loop may be inoperable for up to 2 hours for surveillance testing provided the other loop is OPERABLE and in operation.

*The shutdown cooling pump may be removed from operation for up to 2 hours per 8 hour period provided the other loop is OPERABLE.

(b) ##The RHR shutdown cooling mode loop may be removed from operation during hydrostatic testing.