UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 140
License No. DPR-19

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated September 17, 1993, as supplemented by letter dated June 30, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-19 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 140, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented no later than December 31, 1995.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stang, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 21. 1995

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-249

DRESDEN NUCLEAR POWER STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 134
License No. DPR-25

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated September 17, 1993, as supplemented by letter dated June 30, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 3.B. of Facility Operating License No. DPR-25 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 134, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented no later than December 31, 1995.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stang, Senior Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: September 21, 1995

ATTACHMENT TO LICENSE AMENDMENT NOS. 140 AND 134

FACILITY OPERATING LICENSE NOS. DPR-19 AND DPR-25

DOCKET NOS. 50-237 AND 50-249

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number.

UNIT 2 REMOVE	UNIT 3 REMOVE	INSERT
3/4.6-1	3/4.6-1	3/4.6-1
3/4.6-2	3/4.6-2	3/4.6-2
3/4.6-3	3/4.6-3	3/4.6-3
3/4.6-4	3/4.6-4	3/4.6-4
3/4.6-5	3/4.6-5	3/4.6-5
3/4.6-6	3/4.6-6	3/4.6-6
3/4.6-7	3/4.6-7	3/4.6-7
3/4.6-8	3/4.6-8	3/4.6-8
3/4.6-9	3/4.6-9	3/4.6-9
3/4.6-10	3/4.6-10	3/4.6-10
3/4.6-11	3/4.6-11	3/4.6-11
3/4.6-12	3/4.6-12	3/4.6-12
3/4.6-13	3/4.6-13	3/4.6-13
3/4.6-14	3/4.6-14	3/4.6-14
3/4.6-15	3/4.6-15	3/4.6-15
3/4.6-16	3/4.6-16	3/4.6-16
3/4.6-17	3/4.6-17	3/4.6-17
3/4.6-18	3/4.6-18	3/4.6-18
3/4.6-19	3/4.6-19	3/4.6-19
3/4.6-20	3/4.6-20	3/4.6-20
3/4.6-21	3/4.6-21	3/4.6-21
3/4.6-22	3/4.6-22	3/4.6-22
3/4.6-23	3/4.6-23	3/4.6-23
3/4.6-24	3/4.6-24	3/4.6-24
---	---	3/4.6-25
---	---	3/4.6-26
---	---	3/4.6-27
B 3/4.6-25	B 3/4.6-25	B 3/4.6-1
B 3/4.6-26	B 3/4.6-26	B 3/4.6-2
B 3/4.6-26a	B 3/4.6-26a	B 3/4.6-3
B 3/4.6-27	B 3/4.6-27	B 3/4.6-4
B 3/4.6-28	B 3/4.6-28	B 3/4.6-5
B 3/4.6-29	B 3/4.6-29	B 3/4.6-6
B 3/4.6-30	B 3/4.6-30	B 3/4.6-7
B 3/4.6-31	B 3/4.6-31	B 3/4.6-8
B 3/4.6-32	B 3/4.6-32	B 3/4.6-9
B 3/4.6-33	B 3/4.6-33	---
B 3/4.6-34	B 3/4.6-34	---
B 3/4.6-35	B 3/4.6-35	---
B 3/4.6-36	B 3/4.6-36	---
B 3/4.6-37	B 3/4.6-37	---
B 3/4.6-38	B 3/4.6-38	---
B 3/4.6-39	B 3/4.6-39	---

3.6 - LIMITING CONDITIONS FOR OPERATION

A. Recirculation Loops

Two reactor coolant system recirculation loops shall be in operation.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

1. With only one reactor coolant system recirculation loop in operation, 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 recirculation loop operation per Specifications 2.2.A and 3.2.E.

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

4.6 - SURVEILLANCE REQUIREMENTS

A. Recirculation Loops

Each pump motor generator (MG) set scoop tube mechanical and electrical stop shall be demonstrated OPERABLE with the overspeed setpoints specified in the CORE OPERATING LIMITS REPORT at least once per 18 months.

3.6 - LIMITING CONDITIONS FOR OPERATION

4.6 - SURVEILLANCE REQUIREMENTS

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

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

2. With no reactor coolant system recirculation loops in operation, immediately initiate measures to place the unit in at least STARTUP within 8 hours and in HOT SHUTDOWN within the next 6 hours.

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

3.6 - LIMITING CONDITIONS FOR OPERATION

B. Jet Pumps

All jet pumps shall be OPERABLE and flow indication shall be OPERABLE on at least 18 jet pumps[a].

APPLICABILITY:

OPERATIONAL 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. INTENTIONALLY LEFT BLANK.
3. With flow indication inoperable for both jet pumps on 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 on 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.

4.6 - SURVEILLANCE REQUIREMENTS

B. Jet Pumps

All jet pumps shall be demonstrated OPERABLE as follows:

1. During two loop operation, at least once per 24 hours while greater than 25% of RATED THERMAL POWER by determining recirculation loop flow, total core flow and individual jet pump flow for each jet pump and verifying that no two of the following conditions occur when both recirculation pumps are operating in accordance with Specification 3.6.C:
   a. The indicated recirculation pump flow differs by >10% from the established speed-flow characteristics.
   b. The indicated total core flow differs by >10% from the established total core flow value derived from established core plate ΔP/core flow relationships.
   c. The indicated flow of any individual jet pump differs from the established patterns by >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.

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a Inoperable flow indication shall not be allowed on both jet pumps sharing a jet pump riser, nor on both calibrated jet pumps on the same recirculation loop.

3.6 - LIMITING CONDITIONS FOR OPERATION

4.6 - SURVEILLANCE REQUIREMENTS

2. During single recirculation loop operation, at least once per 24 hours while greater than 25% of RATED THERMAL POWER by verifying that no two of the following conditions occur:

   a. The indicated recirculation pump flow in the operating loop differs by >10% from the established single recirculation speed-flow characteristics.

   b. The indicated total core flow differs by >10% from the established total core flow value derived from established core plate ΔP/core flow relationships.

   c. The indicated flow of any individual jet pump differs from established single recirculation loop patterns by >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.

3.6 - LIMITING CONDITIONS FOR OPERATION

C. Recirculation Pumps

Recirculation pump speed shall be maintained within:

1. 10% of each other with THERMAL POWER ≥80% of RATED THERMAL POWER.

2. 15% of each other with THERMAL POWER <80% of RATED THERMAL POWER.

APPLICABILITY:

OPERATIONAL 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. Restore the recirculation pump speeds to within the specified limit within 2 hours, or

2. Trip one of the recirculation pumps and take the ACTION required by Specification 3.6.A.1.

4.6 - SURVEILLANCE REQUIREMENTS

C. Recirculation Pumps

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

3.6 - LIMITING CONDITIONS FOR OPERATION

D. Idle Recirculation Loop Startup

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 ≤145°F[a], and:

1. 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 ≤50°F, or
2. When only one loop has been idle, unless the temperature differential between the reactor coolant within the idle and operating recirculation loops is ≤50°F and the speed of the operating pump is ≤43% of rated pump speed.

APPLICABILITY:

OPERATIONAL 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.

4.6 - SURVEILLANCE REQUIREMENTS

D. Idle Recirculation Loop Startup

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.

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

3.6 - LIMITING CONDITIONS FOR OPERATION

E. Safety Valves

The safety valve function of the 9 reactor coolant system safety valves shall be OPERABLE in accordance with the specified code safety valve function lift settings[a] established as:

1 safety valve[b] @1135 psig ±1%
2 safety valves @1240 psig ±1%
2 safety valves @1250 psig ±1%
4 safety valves @1260 psig ±1%

Each installed safety valve shall be closed with OPERABLE position indication.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With the safety valve function of one or more of the above required safety valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 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.

4.6 - SURVEILLANCE REQUIREMENTS

E. Safety Valves

1. The position indicators for each safety 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.

2. At least once per 18 months, 1/2 of the safety 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 40 months, the safety valves shall be rotated such that all 9 safety 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.

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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 valve.

3.6 - LIMITING CONDITIONS FOR OPERATION

F. Relief Valves

5 reactor coolant system relief valves and the reactuation time delay of two relief valves shall be OPERABLE with the following settings:

Relief Function
Setpoint (psig)

Open
≤ 1112 psig
≤ 1112 psig
≤ 1135 psig
≤ 1135 psig
≤ 1135 psig[a]

Each installed relief valve shall be closed with OPERABLE position indication.

APPLICABILITY:

OPERATIONAL 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.

4.6 - SURVEILLANCE REQUIREMENTS

F. Relief Valves

1. The relief valve function and the reactuation time delay function instrumentation shall be demonstrated OPERABLE by performance of a:
   a. INTENTIONALLY LEFT BLANK
   b. CHANNEL CALIBRATION, LOGIC SYSTEM FUNCTIONAL TEST and simulated automatic operation of the entire system at least once per 18 months.
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.

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a Target Rock combination safety/relief valve.

3.6 - LIMITING CONDITIONS FOR OPERATION

2. With the relief valve function and/or the reactuation time delay of one of the above required reactor coolant system relief valves inoperable, restore the inoperable relief valve function and the reactuation time delay 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. With the relief valve function and/or the reactuation time delay of more than one of the above required reactor coolant system 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.

4.6 - SURVEILLANCE REQUIREMENTS

3.6 - LIMITING CONDITIONS FOR OPERATION

G. Leakage Detection Systems

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

1. The primary containment atmosphere particulate radioactivity sampling system, and
2. The drywell floor drain sump system.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With the primary containment atmosphere particulate radioactivity sampling 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 sump 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.

4.6 - SURVEILLANCE REQUIREMENTS

G. Leakage Detection Systems

The reactor coolant system leakage detection systems shall be demonstrated OPERABLE by:

1. Performing the leakage determinations 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.

3.6 - LIMITING CONDITIONS FOR OPERATION

H. Operational Leakage

Reactor coolant system leakage shall be limited to:

1. No PRESSURE BOUNDARY LEAKAGE.
2. ≤25 gpm total leakage averaged over any 24 hour surveillance period.
3. ≤5 gpm UNIDENTIFIED LEAKAGE.
4. ≤2 gpm increase in UNIDENTIFIED LEAKAGE within any period of 24 hours or less (Applicable in OPERATIONAL MODE 1 only).

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.
2. With the reactor coolant system UNIDENTIFIED LEAKAGE or total leakage rate(s) greater than the above limit(s), 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.

4.6 - SURVEILLANCE REQUIREMENTS

H. Operational Leakage

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

1. Sampling the primary containment atmospheric particulate radioactivity at least once per 12 hours[a], and
2. Determining the primary containment sump flow rate at least once per 8 hours, not to exceed 12 hours.

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a Not a means of quantifying leakage.

3.6 - LIMITING CONDITIONS FOR OPERATION | 4.6 - SURVEILLANCE REQUIREMENTS

3. With an increase in reactor coolant system UNIDENTIFIED LEAKAGE of >2 gpm within any period of 24 hours or less in OPERATIONAL MODE 1:

   a. Identify the source of leakage as not IGSCC susceptible material within 4 hours, or

   b. Be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3.6 - LIMITING CONDITIONS FOR OPERATION

I. Chemistry

The chemistry of the reactor coolant system shall be maintained within the limits specified in Table 3.6.I-1.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2[a] and 3[a].

ACTION:

1. In OPERATIONAL MODE 1:

   a. With the conductivity, chloride concentration or pH exceeding the limit specified in Table 3.6.I-1;

      1) For ≤72 hours during one continuous time interval, and

      2) For ≤336 hours per year for conductivity and chloride concentration, and

      3) With the conductivity ≤10 μmho/cm at 25°C and with the chloride concentration ≤0.5 ppm,

      the condition does not need to be reported to the Commission.

   b. With the conductivity, chloride concentration or pH exceeding the limit specified in Table 3.6.I-1;

      1) For >72 hours during one continuous time interval, or

4.6 - SURVEILLANCE REQUIREMENTS

I. Chemistry

The reactor coolant shall be determined to be within the specified chemistry limit by:

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

2. Analyzing a sample of the reactor coolant for:

   a. Chlorides at least once per:

      1) 72 hours, and

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

   b. Conductivity at least once per 72 hours.

   c. pH at least once per 8 hours whenever conductivity is greater than the limit in Table 3.6.I-1.

3. Continuously recording the conductivity of the reactor coolant, or, when the continuous recording conductivity monitor is inoperable, obtaining an in-line conductivity measurement at least once per 4 hours.

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a The provisions of Specification 3.0.D are not applicable during unit shutdown when entering OPERATIONAL MODE(s) 2 and 3 from OPERATIONAL MODE 1.

3.6 - LIMITING CONDITIONS FOR OPERATION

2) For >336 hours per year for conductivity and chloride concentration,

Be in at least STARTUP within the next 8 hours.

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

2. In OPERATIONAL MODE(s) 2 and 3 with the conductivity, chloride concentration or pH exceeding the limit specified in Table 3.6.I-1 for >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.

4.6 - SURVEILLANCE REQUIREMENTS

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

   a. 7 days, and

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

TABLE 3.6.I-1

REACTOR COOLANT SYSTEM CHEMISTRY LIMITS

OPERATIONAL MODE(s)	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

3.6 - LIMITING CONDITIONS FOR OPERATION

J. Specific Activity

The specific activity of the reactor coolant shall be limited to $\leq$0.2 μCi/gram DOSE EQUIVALENT I-131.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. In OPERATIONAL MODE(s) 1, 2 or 3 with the specific activity of the reactor coolant >0.2 μCi/gram DOSE EQUIVALENT I-131 but $\leq$4.0 μCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or >4.0 μCi/gram DOSE EQUIVALENT I-131, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours.

2. In OPERATIONAL MODE(s) 1, 2 or 3, with the specific activity of the reactor coolant >0.2 μCi/gram DOSE EQUIVALENT I-131, perform the sampling and analysis requirements of Item 3.a of Table 4.6.J-1 until the specific activity of the reactor coolant is restored to within its limit.

3. In OPERATIONAL MODE(s) 1 or 2, with:

4.6 - SURVEILLANCE REQUIREMENTS

J. Specific Activity

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 4.6.J-1.

3.6 - LIMITING CONDITIONS FOR OPERATION

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

b. The offgas level, prior to the holdup line, increased by >25,000 μCi/second in one hour during steady state operation at release rates <100,000 μCi/second, or

c. The offgas level, prior to the holdup line, increased by >15% in one hour during steady state operation at release rates >100,000 μCi/second,

Perform the sampling and analysis requirements of Item 3.b of Table 4.6.J-1 until the specific activity of the reactor coolant is restored to within its limit.

4.6 - SURVEILLANCE REQUIREMENTS

TABLE 4.6.J-1

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

Type of Measurement and Analysis	Sample and Analysis Frequency	OPERATIONAL 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. Isotopic Analysis for Iodine	a) At least once per 4 hours, whenever the specific activity exceeds a limit, as required by ACTION 2.	1[a], 2[a], 3[a]
	b) At least one sample, between 2 and 6 hours following the change in THERMAL POWER or off-gas level, as required by ACTION 3.	1[a], 2[a]
4. Isotopic Analysis of an Off-gas Sample Including Quantitative Measurements for Xe-133, Xe-135 and Kr-83	At least once per 31 days	1

a Until the specific activity of the reactor coolant system is restored to within its limits.

3.6 - LIMITING CONDITIONS FOR OPERATION

K. Pressure/Temperature Limits

The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.6.K-1 (1) curve A for hydrostatic or leak testing; (2) curve B for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and (3) curve C for operations with a critical core other than low power PHYSICS TESTS, with:

1. A maximum reactor coolant heatup of 100°F in any one hour period,
2. A maximum reactor coolant cooldown of 100°F in any one hour period,
3. A maximum reactor coolant temperature change of ≤20°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and
4. The reactor vessel flange and head flange temperature ≥100°F when reactor vessel head bolting studs are under tension.

APPLICABILITY:

At all times.

4.6 - SURVEILLANCE REQUIREMENTS

K. Pressure/Temperature Limits

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 required heatup and cooldown limits and to the right of the limit lines of Figure 3.6.K-1 curves A, or B, as applicable, at least once per 30 minutes.
2. The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.6.K-1 curve 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.
3. The reactor vessel material surveillance specimens shall be removed and examined, to determine changes in reactor pressure vessel material properties in accordance with 10CFR Part 50, Appendix H.
4. The reactor vessel flange and head flange temperature shall be verified to be ≥100°F:
   a. In OPERATIONAL MODE 4 when the reactor coolant temperature is:
      1) ≤130°F, at least once per 12 hours.
      2) ≤110°F, at least once per 30 minutes.
   b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.

3.6 - LIMITING CONDITIONS FOR OPERATION

4.6 - SURVEILLANCE REQUIREMENTS

ACTION:

With any of the above limits exceeded,

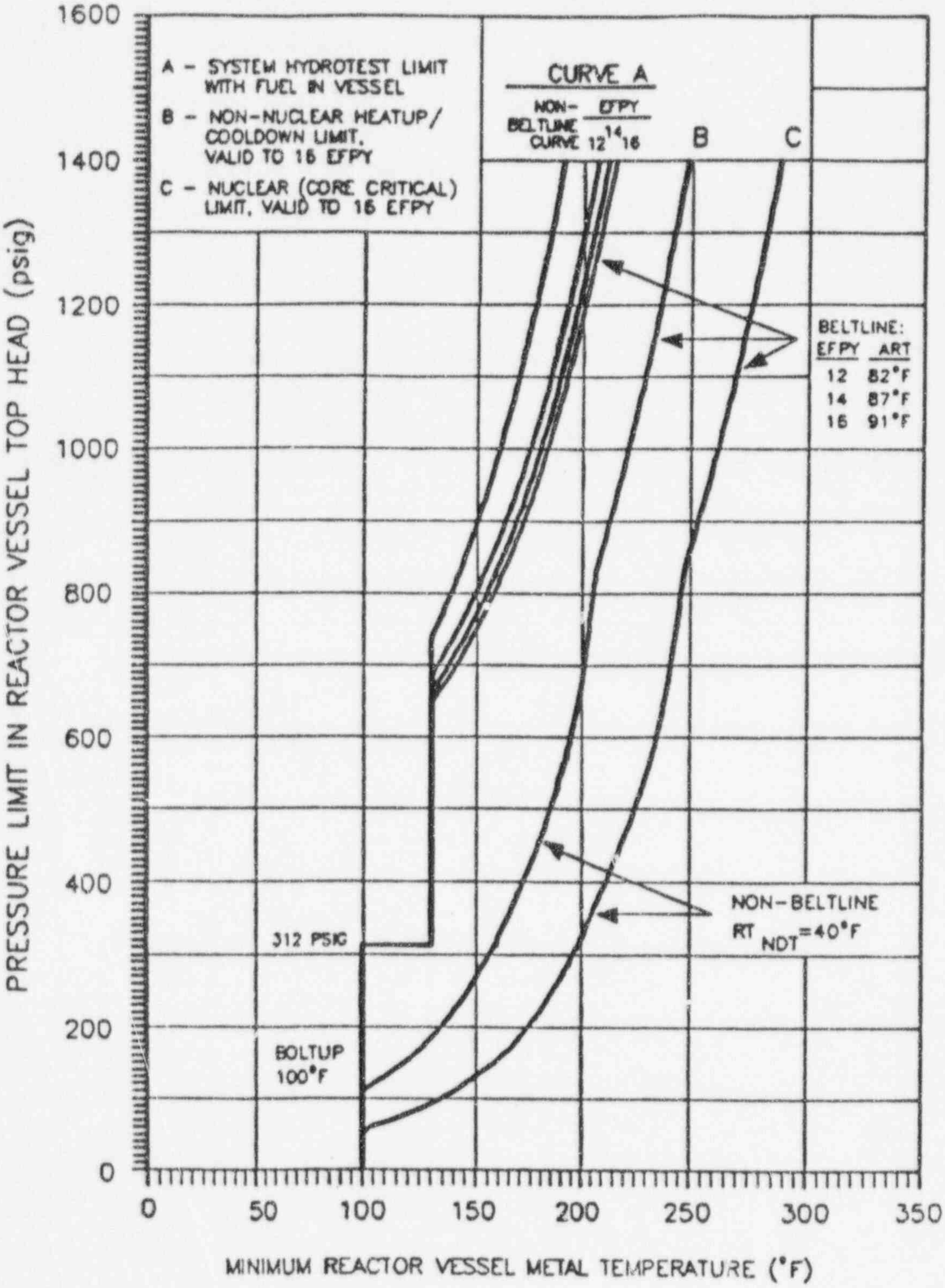
1. Restore the temperature and/or pressure to within the limits within 30 minutes, and

2. Perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system and determine that the reactor coolant system remains acceptable for continued operations, or

3. Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

FIGURE 3.6.K-1

MINIMUM REACTOR VESSEL METAL TEMPERATURE VS. REACTOR VESSEL PRESSURE



3.6 - LIMITING CONDITIONS FOR OPERATION

L. Reactor Steam Dome Pressure

The pressure in the reactor steam dome shall be ≤1005 psig.

APPLICABILITY:

OPERATIONAL MODE(s) 1[a] and 2[a]

ACTION:

With the reactor steam dome pressure >1005 psig, reduce the pressure to ≤1005 psig within 15 minutes or be in at least HOT SHUTDOWN within 12 hours.

4.6 - SURVEILLANCE REQUIREMENTS

L. Reactor Steam Dome Pressure

The reactor steam dome pressure shall be verified to be ≤1005 psig at least once per 12 hours.

a Not applicable during anticipated transients.

3.6 - LIMITING CONDITIONS FOR OPERATION

M. Main Steam Line Isolation Valves

Two main steam line isolation valves (MSIVs) per main steam line shall be OPERABLE with closing times ≥3 seconds and ≤5 seconds.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

With one or more MSIVs inoperable, maintain at least one MSIV OPERABLE in each affected main steam line that is open and within 8 hours either:

1. Restore the inoperable valve(s) to OPERABLE status, or
2. Isolate the affected main steam line by use of a deactivated MSIV in the closed position.

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

4.6 - SURVEILLANCE REQUIREMENTS

M. Main Steam Line Isolation Valves

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.E.

3.6 - LIMITING CONDITIONS FOR OPERATION

N. Structural Integrity

The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.6.N.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, 4 and 5.

ACTION:

1. 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 limits 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. 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).

3. 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.

4.6 - SURVEILLANCE REQUIREMENTS

N. Structural Integrity

No additional Surveillance Requirements other than those required by Specification 4.0.E.

3.6 - LIMITING CONDITIONS FOR OPERATION

O. Shutdown Cooling - HOT SHUTDOWN

Two[a] shutdown cooling (SDC) loops shall be OPERABLE and, unless at least one recirculation pump is in operation, at least one shutdown cooling loop shall be in operation[b][c], with each loop consisting of at least:

1. One OPERABLE SDC pump, and

2. One OPERABLE SDC heat exchanger.

APPLICABILITY:

OPERATIONAL MODE 3, with reactor vessel coolant temperature less than the SDC cut-in permissive setpoint.

ACTION:

1. With less than the above required SDC loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible. Within 1 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 SDC loop. Be in at least COLD SHUTDOWN within 24 hours[d].

4.6 - SURVEILLANCE REQUIREMENTS

O. Shutdown Cooling - HOT SHUTDOWN

At least one SDC loop, one recirculation pump or alternate method shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

---

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

b A 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 loop may be removed from operation during hydrostatic testing.

d Whenever two or more 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.

3.6 - LIMITING CONDITIONS FOR OPERATION

4.6 - SURVEILLANCE REQUIREMENTS

2. With no SDC loop or recirculation pump in operation, immediately initiate corrective action to return at least one shutdown cooling loop or recirculation pump to operation as soon as possible. Within 1 hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

3.6 - LIMITING CONDITIONS FOR OPERATION

P. Shutdown Cooling - COLD SHUTDOWN

Two[a] shutdown cooling (SDC) loops shall be OPERABLE and, unless at least one recirculation pump is in operation, at least one shutdown cooling loop shall be in operation[b][c] with each loop consisting of at least:

1. One OPERABLE SDC pump, and

2. One OPERABLE SDC heat exchanger.

APPLICABILITY:

OPERATIONAL MODE 4.

ACTION:

1. With less than the above required SDC loops OPERABLE, within 1 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 SDC loop.

2. With no SDC loop or recirculation pump in operation, within 1 hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

4.6 - SURVEILLANCE REQUIREMENTS

P. Shutdown Cooling - COLD SHUTDOWN

At least one SDC loop, recirculation pump or alternate method shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

---

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

b A 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 loop may be removed from operation during hydrostatic testing.

BASES

3/4.6.A Recirculation Loops

3/4.6.B Jet Pumps

3/4.6.C Recirculation Pumps

3/4.6.D Idle Recirculation Loop Startup

The reactor coolant recirculation system is designed to provide a forced coolant flow through the core to remove heat from the fuel. The reactor coolant recirculation system consists of two recirculation pump loops external to the reactor vessel. These loops provide the piping path for the driving flow of water to the reactor vessel jet pumps. The operation of the reactor coolant recirculation system is an initial condition assumed in the design basis loss-of-coolant accident (LOCA). During a LOCA caused by a recirculation loop pipe break, the intact loop is assumed to provide coolant flow during the first few seconds of the accident. The analyses assumes both loops are operating at the same flow prior to the accident. If a LOCA occurs with a flow mismatch between the two loops, the analysis conservatively assumes the pipe break is in the loop with the higher flow.

A plant specific analysis has been performed assuming only one operating recirculation loop. This analysis has demonstrated that in the event of a LOCA caused by a pipe break in the operating recirculation loop, the ECCS response will provide adequate core cooling. The transient analyses of Chapter 15 of the FSAR have also been performed for single recirculation loop operation and demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed provided the MCPR fuel cladding integrity Safety Limit is increased as noted by Specification 2.1.B. The Reactor Protection System APRM scram and control rod block setpoints are also required to be adjusted to account for the different response of the reactor and different relationships between recirculation drive flow and reactor core flow. During single loop operation for greater than 24 hours, the idle recirculation pump is electrically prohibited from starting until ready to resume two loop operation. This is done to prevent a cold water injection transient caused by an inadvertent pump startup.

Jet pump OPERABILITY is an explicit assumption in the design basis LOCA analysis. The capability of reflooding the core to two-thirds core height is dependent upon the structural integrity of the jet pumps. If a beam holding a jet pump in place fails, the jet pump suction and mixer sections could become displaced, resulting in a larger flow area through the jet pump and a lower core flooding elevation. This could adversely affect the water level in the core during the reflood phase of a LOCA as well as the assumed blowdown flow during a LOCA.

The surveillance requirements for jet pumps are designed to detect a significant degradation in jet pump performance that precedes a jet pump failure. Significant degradation is indicated if more than one of the three specified criteria confirms unacceptable deviations from established patterns or relationships. A break in a jet pump decreases the flow resistance characteristic of the external piping loop causing the recirculation pump to operate at a higher flow condition when compared to previous operation. The agreement of indicated core plate dp and core flow relationships provides

BASES

assurance that the recirculation flow is not bypassing the core through inactive or broken jet pumps. The change in the flow rate of the failed jet pump produces a change in the indicated flow rate of that pump relative to the other pumps in that loop. Comparison of the data with a normal relationship or pattern provides the indication necessary to detect a failed jet pump.

The accuracy of the core flow measurement system is assumed in the derivation of the Safety Limit MINIMUM CRITICAL POWER RATIO. An analysis assuming a loss of flow indication for three jet pumps resulted in uncertainties within the values assumed for the core flow measurement system in the Safety Limit MINIMUM CRITICAL POWER RATIO calculation for both two loop operation and single loop operation. Therefore, plant operation with loss of flow indication in up to two jet pumps is acceptable as long as each jet pump is on a separate riser and no more than one calibrated double tap jet pump per loop is affected.

Recirculation pump speed mismatch limits are in compliance with the ECCS LOCA analysis design criteria. For some limited low probability events with the recirculation loop operating with large speed differences, it is possible for the LPCI loop selection logic to select the wrong loop for injection. Above 80% of RATED THERMAL POWER, the LPCI selection logic is expected to function at a speed differential of 15%. Below 80% of RATED THERMAL POWER, the loop select logic would be expected to function at a speed differential of 20%. Therefore, this specification provides a margin of 5% in pump speed differential before a problem could arise.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. The loop temperature must also be within 50°F of the reactor pressure vessel steam space coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles. Since the coolant in the bottom of the vessel is at a lower temperature than the coolant in the upper regions of the core, undue stress on the vessel would result if the temperature difference was greater than 145°F. Additionally, asymmetric speed operation of the recirculation pumps during idle loop startup induces levels of jet pump riser vibration that are higher than normal. The specific limitation of 43% of rated pump speed for the operating recirculation pump prior to the start of the idle recirculation pump ensures that the recirculation pump speed mismatch requirements are maintained.

3/4.6.E Safety Valves

3/4.6.F Relief Valves

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of safety valves are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary. The overpressure protection system must accommodate the most severe pressurization transient. Evaluations have determined that the most severe transient is the closure of all the main steam line isolation valves followed by a reactor scram on high neutron flux. The analysis results demonstrate that the design safety valve capacity is capable of maintaining reactor pressure below the ASME Code limit of 110% of the reactor pressure vessel design pressure.

The relief valve function is not assumed to operate in response to any accident, but are provided to remove the generated steam flow upon turbine stop valve closure coincident with failure of the turbine bypass system. The relief valve opening pressure settings are sufficiently low to prevent the need for safety valve actuation following such a transient.

Each of the five relief valves discharge to the suppression chamber via a dedicated relief valve discharge line. Steam remaining in the relief valve discharge line following closure can condense, creating a vacuum which may draw suppression pool water up into the discharge line. This condition is normally alleviated by the vacuum breakers; however, subsequent actuation in the presence of an elevated water leg can result in unacceptably high thrust loads on the discharge piping. To prevent this, the relief valves have been designed to ensure that each valve which closes will remain closed until the normal water level in the relief valve discharge line is restored. The opening and closing setpoints are set such that all pressure induced subsequent actuation are limited to the two lowest set valves. These two valves are equipped with additional logic which functions in conjunction with the setpoints to inhibit valve reopening during the elevated water leg duration time following each closure.

Each safety/relief valve is equipped with diverse position indicators which monitor the tailpipe acoustic vibration and temperature. Either of these provide sufficient indication of safety/relief valve position for normal operation.

3/4.6.G Leakage Detection Systems

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. Limits on leakage from the reactor coolant pressure boundary are required so that appropriate action can be taken before the integrity of the reactor coolant pressure boundary is impaired. Leakage detection systems for the reactor coolant system are provided to alert the operators when leakage rates above the normal background levels are detected and also to supply quantitative measurement of leakage rates. Leakage from the reactor coolant pressure boundary inside the drywell is detected by at least one or two independently monitored variables, such as sump level changes and drywell atmosphere radioactivity levels. The means of quantifying leakage in the drywell is the drywell floor drain sump pumps. With the drywell floor drain sump pump system inoperable, no other form of monitoring can provide the equivalent information. However, primary containment atmosphere sampling for radioactivity can provide indication of changes in leakage rates.

BASES

3/4.6.H Operational Leakage

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shutdown to allow further investigation and corrective action.

An UNIDENTIFIED LEAKAGE increase of more than 2 gpm within a 24 hour period is an indication of a potential flaw in the reactor coolant pressure boundary and must be quickly evaluated. Although the increase does not necessarily violate the absolute UNIDENTIFIED LEAKAGE limit, IGSCC susceptible components must be determined not to be the source of the leakage within the required completion time.

3/4.6.I Chemistry

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low, thus the 0.2 ppm limit on chlorides is permitted during POWER OPERATION.

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity meter inoperable, additional samples must be analyzed to ensure that the chlorides are not exceeding the limits.

Action 1 permits temporary operation with chemistry limits outside of the limits required in OPERATIONAL MODE 1 without requiring Commission notification. The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.6.J Specific Activity

The limitations on the specific activity of the primary coolant ensure that the 2 hour thyroid and whole body doses resulting from a main steam line failure outside the containment during steady state operation will not exceed small fractions of the dose guidelines of 10 CFR 100. The values

BASES

3/4.6.J Specific Activity

The limitations on the specific activity of the primary coolant ensure that the 2 hour thyroid and whole body doses resulting from a main steam line failure outside the containment during steady state operation will not exceed small fractions of the dose guidelines of 10 CFR 100. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131, but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analysis following power changes may be permissible if justified by the data obtained.

Closing the main steam line isolation valves prevents the release of activity to the environs should a steam line rupture occur outside containment. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

3/4.6.K Pressure/Temperature Limits

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

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

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

The pressure-temperature limit lines shown in Figure 3.6.K-1, for operating conditions; Inservice Hydrostatic Testing (curve A), Non-Nuclear Heatup/Cooldown (curve B), and Core Critical Operation (curve C). The curves have been established to be in conformance with Appendix G to 10 CFR Part 50 and Regulatory Guide 1.99 Revision 2, and take into account the change in reference nil-ductility transition temperature (RT_{NDT}) as a result of neutron embrittlement. The adjusted reference temperature (ART) of the limiting vessel material is used to account for irradiation effects.

Three vessel regions are considered for the development of the pressure-temperature curves: 1) the core beltline region; 2) the non-beltline region (other than the closure flange region); and 3) the closure flange region. The beltline region is defined as that region of the reactor vessel that directly surrounds the effective height of the reactor core and is subject to an RT_{NDT} adjustment to account for radiation embrittlement. The non-beltline and closure flange regions receive insufficient fluence to necessitate an RT_{NDT} adjustment. These regions contain components which include; the reactor vessel nozzles, closure flanges, top and bottom head plates, control rod drive penetrations, and shell plates that do not directly surround the reactor core. Although the closure flange region is a non-beltline region, it is treated separately for the development of the pressure-temperature curves to address 10CFR Part 50 Appendix G requirements.

In evaluating the adequacy of the steel which comprises the reactor vessel, it is necessary that the following be established: 1) the RT_{NDT} for all vessel and adjoining materials; 2) the relationship between RT_{NDT} and integrated neutron flux (fluence, at energies greater than one Mev); and 3) the fluence at the location of a postulated flaw.

<u>Boltup Temperature</u>

The initial RT_{NDT} of the main closure flanges, the shell and head materials connecting to these flanges, and connecting welds is 10°F; however, the vertical electroslag welds which terminate immediately below the vessel flange have an RT_{NDT} of 40°F. Therefore, the minimum allowable boltup temperature is established as 100°F (RT_{NDT} + 60°F) which includes a 60°F conservatism required by the original ASME Code of construction.

<u>Curve A - Hydrotesting</u>

As indicated in curve A of Figure 3.6.K-1 for system hydrotesting, the minimum metal temperature of the reactor vessel shell is 100°F for reactor pressures less than 312 psig. This 100°F minimum boltup temperature is based on a RT_{NDT} of 40°F for the electroslag weld immediately below the vessel flange and a 60°F conservatism required by the original ASME Code of construction. At reactor pressures greater than 312 psig, the minimum vessel metal temperature is established as 130°F. The 130°F minimum temperature is based on a closure flange region RT_{NDT} of 40°F and a 90°F conservatism required by 10CFR Part 50 Appendix G for pressure in excess of 20% of the preservice hydrostatic test pressure (1563 psig). At approximately 650 psig the effects of pressurization are more limiting than the boltup stresses at the closure flange region, hence a family of non-linear curves intersect the 130°F vertical line. Beltline as well as non-beltline curves have been provided to allow separate monitoring of the two regions. Beltline curves as a function of vessel exposure for 12, 14 and 16 effective full power years (EFPY) are presented to allow the use of the appropriate curve up to 16 EFPY of operation.

A typical sequence involved in pressure testing is a heatup to the required temperature and then pressurization to the required pressure for the inspection. During the heatup, at 100°F/hour or less, Curve B is the governing curve. Since the vessel is not pressurized during the heatup, Curves A and B are the same. When temperatures are stabilized to within 20°F/hour rates, at temperatures above those required by curve A, pressurization begins, at which point Curve A is the governing curve. During the inspection period with the vessel at the required pressure, temperature changes are limited to 20°F/hour.

Curve B - Non-Nuclear Heatup/Cooldown

Curve B of Figure 3.6.K-1 applies during heatups with non-nuclear heat (e.g., recirculation pump heat) and during cooldowns when the reactor is not critical (e.g., following a scram). The curve provides the minimum reactor vessel metal temperatures based on the most limiting vessel stress. As indicated by the vertical 100°F line, the boltup stresses at the closure flange region are most limiting for reactor pressures below approximately 110 psig. For reactor pressures greater than approximately 110 psig, pressurization and thermal stresses become more limiting than the boltup stresses, which is reflected by the nonlinear portion of curve B. The non-linear portion of the curve is dependent on non-beltline and beltline regions, with the beltline region temperature limits having been adjusted to account for vessel irradiation (up to a vessel exposure of 16 EFPY). The non-beltline region is limiting between approximately 110 psig and 830 psig. Above approximately 803 psig, the beltline region becomes limiting.

Curve C - Core Critical Operation

Curve C, the core critical operation curve shown in Figure 3.6.K-1, is generated in accordance with 10CFR Part 50 Appendix G which requires core critical pressure-temperature limits to be 40°F above any curve A or B limits. Since curve B is more limiting, (curve C is curve B plus 40°F.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTI E185-73 and 10CFR Part 50, Appendix H, irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel in the core area. The irradiated specimens can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figure 3.6.J-1 shall be adjusted, as required, on the basis of the specimen data and recommendations of Regulatory Guide 1.99, Revision 2.

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3/4.6.L Reactor Steam Dome

The reactor steam dome pressure is an assumed initial condition of Design Basis Accidents and transients and is also an assumed value in the determination of compliance with reactor pressure vessel overpressure protection criteria. The reactor steam dome pressure of ≤1005 psig is an initial condition of the vessel overpressure protection analysis. This analysis assumes an initial maximum reactor steam dome pressure and evaluates the response of the pressure relief system, primarily the safety valves, during the limiting pressurization transient. The determination of compliance with the overpressure criteria is dependent on the initial reactor steam dome pressure; therefore, the limit on this pressure ensures that the assumptions of the overpressure protection analysis are conserved.

3/4.6.M Main Steam Line Isolation Valves

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

3/4.6.N Structural Integrity

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

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

3/4.6.O Shutdown Cooling - HOT SHUTDOWN

3/4.6.P Shutdown Cooling - COLD SHUTDOWN

Irradiated fuel in the reactor pressure vessel generates decay heat during normal and abnormal shutdown conditions, potentially resulting in an increase in the temperature of the reactor coolant. This decay heat is required to be removed such that the reactor coolant temperature can be reduced in preparation for performing refueling, maintenance operations or for maintaining the reactor in cold shutdown conditions. Systems capable of removing decay heat are therefore required to perform these functions.

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

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

AND

MIDAMERICAN ENERGY COMPANY

DOCKET NO. 50-254

QUAD CITIES NUCLEAR POWER STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 162
License No. DPR-29

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated September 17, 1993, as supplemented by letter dated June 30, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Facility Operating License No. DPR-29 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 162 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented no later than June 30, 1996.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert M. Pulsifer, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 21, 1995

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

COMMONWEALTH EDISON COMPANY

AND

MIDAMERICAN ENERGY COMPANY

DOCKET NO. 50-265

QUAD CITIES NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 158
License No. DPR-30

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated September 17, 1993, as supplemented by letter dated June 30, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Facility Operating License No. DPR-30 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 158 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented no later than June 30, 1996.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert M. Pulsifer, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: September 21, 1995

ATTACHMENT TO LICENSE AMENDMENT NOS. 162 AND 158

FACILITY OPERATING LICENSE NOS. DPR-29 AND DPR-30

DOCKET NOS. 50-254 AND 50-265

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number.

UNIT 1 REMOVE	UNIT 2 REMOVE	INSERT
3.6/4.6-1	3.6/4.6-1	3/4.6-1
3.6/4.6-2	3.6/4.6-2	3/4.6-2
3.6/4.6-3	3.6/4.6-2a	3/4.6-3
3.6/4.6-4	3.6/4.6-3	3/4.6-4
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3.6/4.6-15d	3.6/4.6-8	3/4.6-20
3.6/4.6-16	3.6/4.6-9	3/4.6-21
3.6/4.6-17	3.6/4.6-9a	3/4.6-22
3.6-4.6-17a	3.6/4.6-10	3/4.6-23
3.6/4.6-18	3.6/4.6-11	3/4.6-24
3.6.4.6-19	3.6/4.6-12	3/4.6-25
3.6/4.6-20	3.6/4.6-12a	3/4.6-26
3.6/4.6-21	3.6/4.6-13	3/4.6-27
3.6/4.6-22	3.6/4.6-13a	3/4.6-28
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3.6/4.6-27	3.6/4.6-18	B 3/4.6-6
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3.6/4.6-30	3.6/4.6-21	B 3/4.6-9
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3.6/4.6-33	Figure 4.6-1	---
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Figure 3.6-1	---	---
Figure 4.6-1	---	---
Figure 4.6-2	---	---

3.6 - LIMITING CONDITIONS FOR OPERATION

A. Recirculation Loops

Two reactor coolant system recirculation loops shall be in operation.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

1. With only one reactor coolant system recirculation loop in operation, 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 recirculation loop operation per Specifications 2.2.A and 3.2.E.

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

4.6 - SURVEILLANCE REQUIREMENTS

A. Recirculation Loops

Each pump motor generator (MG) set scoop tube mechanical and electrical stop shall be demonstrated OPERABLE with the overspeed setpoints specified in the CORE OPERATING LIMITS REPORT at least once per 18 months.

3.6 - LIMITING CONDITIONS FOR OPERATION

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

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

2. With no reactor coolant system recirculation loops in operation, immediately initiate measures to place the unit in at least STARTUP within 8 hours and in HOT SHUTDOWN within the next 6 hours.

4.6 - SURVEILLANCE REQUIREMENTS

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

3.6 - LIMITING CONDITIONS FOR OPERATION

B. Jet Pumps

All jet pumps shall be OPERABLE and flow indication shall be OPERABLE on at least 18 jet pumps[a].

APPLICABILITY:

OPERATIONAL 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 on 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 on 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.

4.6 - SURVEILLANCE REQUIREMENTS

B. Jet Pumps

All jet pumps shall be demonstrated OPERABLE as follows:

1. During two loop operation, at least once per 24 hours while greater than 25% of RATED THERMAL POWER by determining recirculation loop flow, total core flow and individual jet pump flow for each jet pump and verifying that no two of the following conditions occur when both recirculation pumps are operating in accordance with Specification 3.6.C:
   a. The indicated recirculation pump flow differs by >10% from the established speed-flow characteristics.
   b. The indicated total core flow differs by >10% from the established total core flow value derived from established core plate ΔP/core flow relationships.
   c. The indicated flow of any individual jet pump differs from the established patterns by >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.

---

a Inoperable flow indication shall not be allowed on both jet pumps sharing a jet pump riser, nor on both calibrated jet pumps on the same recirculation loop.

3.6 - LIMITING CONDITIONS FOR OPERATION

4.6 - SURVEILLANCE REQUIREMENTS

2. During single recirculation loop operation, at least once per 24 hours while greater than 25% of RATED THERMAL POWER by verifying that no two of the following conditions occur:

    a. The indicated recirculation pump flow in the operating loop differs by >10% from the established single recirculation speed-flow characteristics.

    b. The indicated total core flow differs by >10% from the established total core flow value derived from established core plate ΔP/core flow relationships.

    c. The indicated flow of any individual jet pump differs from established single recirculation loop patterns by >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.

3.6 - LIMITING CONDITIONS FOR OPERATION

C. Recirculation Pumps

Recirculation pump speed shall be maintained within:

1. 10% of each other with THERMAL POWER ≥80% of RATED THERMAL POWER.

2. 15% of each other with THERMAL POWER <80% of RATED THERMAL POWER.

APPLICABILITY:

OPERATIONAL 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. Restore the recirculation pump speeds to within the specified limit within 2 hours, or

2. Trip one of the recirculation pumps and take the ACTION required by Specification 3.6.A.1.

4.6 - SURVEILLANCE REQUIREMENTS

C. Recirculation Pumps

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

3.6 - LIMITING CONDITIONS FOR OPERATION

D. Idle Recirculation Loop Startup

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 ≤145°F[a], and:

1. 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 ≤50°F, or

2. When only one loop has been idle, unless the temperature differential between the reactor coolant within the idle and operating recirculation loops is ≤50°F and the speed of the operating pump is ≤45% of rated pump speed.

APPLICABILITY:

OPERATIONAL 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.

4.6 - SURVEILLANCE REQUIREMENTS

D. Idle Recirculation Loop Startup

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.

3.6 - LIMITING CONDITIONS FOR OPERATION

E. Safety Valves

The safety valve function of the 9 reactor coolant system safety valves shall be OPERABLE in accordance with the specified code safety valve function lift settings[a] established as:

1 safety valve[b] @1135 psig ±1%
2 safety valves @1240 psig ±1%
2 safety valves @1250 psig ±1%
4 safety valves @1260 psig ±1%

Each installed safety valve shall be closed with OPERABLE position indication.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With the safety valve function of one or more of the above required safety valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 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.

4.6 - SURVEILLANCE REQUIREMENTS

E. Safety Valves

1. The position indicators for each safety 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.

2. At least once per 18 months, 1/2 of the safety 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 40 months, the safety valves shall be rotated such that all 9 safety 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.

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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 valve.

3.6 - LIMITING CONDITIONS FOR OPERATION

F. Relief Valves

5 reactor coolant system relief valves and the reactuation time delay of two relief valves shall be OPERABLE with the following settings:

Relief Function
Setpoint (psig)

Open
≤1115 psig
≤1115 psig
≤1135 psig
≤1135 psig
≤1135 psig[a]

Each installed relief valve shall be closed with OPERABLE position indication.

APPLICABILITY:

OPERATIONAL 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.

4.6 - SURVEILLANCE REQUIREMENTS

F. Relief Valves

1. The relief valve function and the reactuation time delay function instrumentation shall be demonstrated OPERABLE by performance of a:
    a. INTENTIONALLY LEFT BLANK
    b. CHANNEL CALIBRATION, LOGIC SYSTEM FUNCTIONAL TEST and simulated automatic operation of the entire system at least once per 18 months.
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.

---

a Target Rock combination safety/relief valve.

3.6 - LIMITING CONDITIONS FOR OPERATION

2. With the relief valve function and/or the reactuation time delay of one of the above required reactor coolant system relief valves inoperable, restore the inoperable relief valve function and the reactuation time delay 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. With the relief valve function and/or the reactuation time delay of more than one of the above required reactor coolant system 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.

4.6 - SURVEILLANCE REQUIREMENTS

3.6 - LIMITING CONDITIONS FOR OPERATION

G. Leakage Detection Systems

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

1. The primary containment atmosphere particulate radioactivity sampling system, and

2. The drywell floor drain sump system.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With the primary containment atmosphere particulate radioactivity sampling 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 sump 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.

4.6 - SURVEILLANCE REQUIREMENTS

G. Leakage Detection Systems

The reactor coolant system leakage detection systems shall be demonstrated OPERABLE by:

1. Performing the leakage determinations of Specification 4.6.H.

2. Performing a CHANNEL CALIBRATION of the drywell floor drain sump pump discharge flow totalizer at least once per 18 months.

3.6 - LIMITING CONDITIONS FOR OPERATION

H. Operational Leakage

Reactor coolant system leakage shall be limited to:

1. No PRESSURE BOUNDARY LEAKAGE.
2. ≤25 gpm total leakage averaged over any 24 hour surveillance period.
3. ≤5 gpm UNIDENTIFIED LEAKAGE.
4. ≤2 gpm increase in UNIDFNTIFIED LEAKAGE within any period of 24 hours or less (Applicable in OPERATIONAL MODE 1 only).

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.
2. With the reactor coolant system UNIDENTIFIED LEAKAGE or total leakage rate(s) greater than the above limit(s), 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.

4.6 - SURVEILLANCE REQUIREMENTS

H. Operational Leakage

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

1. Sampling the primary containment atmospheric particulate radioactivity at least once per 12 hours[a], and
2. Determining the primary containment sump flow rate at least once per 8 hours, not to exceed 12 hours.

a Not a means of quantifying leakage.

3.6 - LIMITING CONDITIONS FOR OPERATION

3. With an increase in reactor coolant system UNIDENTIFIED LEAKAGE of >2 gpm within any period of 24 hours or less in OPERATIONAL MODE 1:

   a. Identify the source of leakage as not IGSCC susceptible material within 4 hours, or

   b. Be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.6 - SURVEILLANCE REQUIREMENTS

3.6 - LIMITING CONDITIONS FOR OPERATION

I. Chemistry

The chemistry of the reactor coolant system shall be maintained within the limits specified in Table 3.6.I-1.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2[a] and 3[a].

ACTION:

1. In OPERATIONAL MODE 1:

   a. With the conductivity, chloride concentration or pH exceeding the limit specified in Table 3.6.I-1;

      1) For ≤72 hours during one continuous time interval, and

      2) For ≤336 hours per year for conductivity and chloride concentration, and

      3) With the conductivity ≤10 μmho/cm at 25°C and with the chloride concentration ≤0.5 ppm,

      the condition does not need to be reported to the Commission.

   b. With the conductivity, chloride concentration or pH exceeding the limit specified in Table 3.6.I-1;

      1) For >72 hours during one continuous time interval, or

4.6 - SURVEILLANCE REQUIREMENTS

I. Chemistry

The reactor coolant shall be determined to be within the specified chemistry limit by:

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

2. Analyzing a sample of the reactor coolant for:

   a. Chlorides at least once per:

      1) 72 hours, and

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

   b. Conductivity at least once per 72 hours.

   c. pH at least once per 8 hours whenever conductivity is greater than the limit in Table 3.6.I-1.

3. Continuously recording the conductivity of the reactor coolant, or, when the continuous recording conductivity monitor is inoperable, obtaining an in-line conductivity measurement at least once per 4 hours.

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a The provisions of Specification 3.0.D are not applicable during unit shutdown when entering OPERATIONAL MODE(s) 2 and 3 from OPERATIONAL MODE 1.

3.6 - LIMITING CONDITIONS FOR OPERATION

2) For >336 hours per year for conductivity and chloride concentration,

Be in at least STARTUP within the next 8 hours.

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

2. In OPERATIONAL MODE(s) 2 and 3 with the conductivity, chloride concentration or pH exceeding the limit specified in Table 3.6.I-1 for >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.

4.6 - SURVEILLANCE REQUIREMENTS

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

   a. 7 days, and

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

TABLE 3.6.I-1

REACTOR COOLANT SYSTEM CHEMISTRY LIMITS

OPERATIONAL MODE(s)	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

(a) Except during chemical decontamination of Reactor Recirculation or Reactor Water Clean-Up system piping.

3.6 - LIMITING CONDITIONS FOR OPERATION

J. Specific Activity

The specific activity of the reactor coolant shall be limited to ≤0.2 μCi/gram DOSE EQUIVALENT I-131.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. In OPERATIONAL MODE(s) 1, 2 or 3 with the specific activity of the reactor coolant >0.2 μCi/gram DOSE EQUIVALENT I-131 but ≤4.0 μCi/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or >4.0 μCi/gram DOSE EQUIVALENT I-131, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours.

2. In OPERATIONAL MODE(s) 1, 2 or 3, with the specific activity of the reactor coolant >0.2 μCi/gram DOSE EQUIVALENT I-131, perform the sampling and analysis requirements of Item 3.a of Table 4.6.J-1 until the specific activity of the reactor coolant is restored to within its limit.

3. In OPERATIONAL MODE(s) 1 or 2, with:

4.6 - SURVEILLANCE REQUIREMENTS

J. Specific Activity

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 4.6.J-1.

3.6 - LIMITING CONDITIONS FOR OPERATION	4.6 - SURVEILLANCE REQUIREMENTS

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

b. The offgas level, prior to the holdup line, increased by >25,000 μCi/second in one hour during steady state operation at release rates <100,000 μCi/second, or

c. The offgas level, prior to the holdup line, increased by >15% in one hour during steady state operation at release rates >100,000 μCi/second,

Perform the sampling and analysis requirements of Item 3.b of Table 4.6.J-1 until the specific activity of the reactor coolant is restored to within its limit.

TABLE 4.6.J-1

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

Type of Measurement and Analysis	Sample and Analysis Frequency	OPERATIONAL 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. Isotopic Analysis for Iodine	a) At least once per 4 hours, whenever the specific activity exceeds a limit, as required by ACTION 2.	1[a], 2[a], 3[a]
	b) At least one sample, between 2 and 6 hours following the change in THERMAL POWER or off-gas level, as required by ACTION 3.	1[a], 2[a]
4. Isotopic Analysis of an Off-gas Sample Including Quantitative Measurements for Xe-133, Xe-135 and Kr-88	At least once per 31 days	1

a Until the specific activity of the reactor coolant system is restored to within its limits.

3.6 - LIMITING CONDITIONS FOR OPERATION

K. Pressure/Temperature Limits

The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.6.K-1 (1) curve A for hydrostatic or leak testing; (2) curve B for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and (3) curve C for operations with a critical core other than low power PHYSICS TESTS, with:

1. A maximum reactor coolant heatup of 100°F in any one hour period,
2. A maximum reactor coolant cooldown of 100°F in any one hour period,
3. A maximum reactor coolant temperature change of ≤20°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and
4. The reactor vessel flange and head flange temperature ≥100°F when reactor vessel head bolting studs are under tension.

APPLICABILITY:

At all times.

4.6 - SURVEILLANCE REQUIREMENTS

K. Pressure/Temperature Limits

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 required heatup and cooldown limits and to the right of the limit lines of Figure 3.6.K-1 curves A, or B, as applicable, at least once per 30 minutes.
2. The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.6.K-1 curve 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.
3. The reactor vessel material surveillance specimens shall be removed and examined, to determine changes in reactor pressure vessel material properties in accordance with 10CFR Part 50, Appendix H.
4. The reactor vessel flange and head flange temperature shall be verified to be ≥100°F:
   a. In OPERATIONAL MODE 4 when the reactor coolant temperature is:
      1) ≤130°F, at least once per 12 hours.
      2) ≤110°F, at least once per 30 minutes.
   b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.

3.6 - LIMITING CONDITIONS FOR OPERATION

4.6 - SURVEILLANCE REQUIREMENTS

ACTION:

With any of the above limits exceeded,

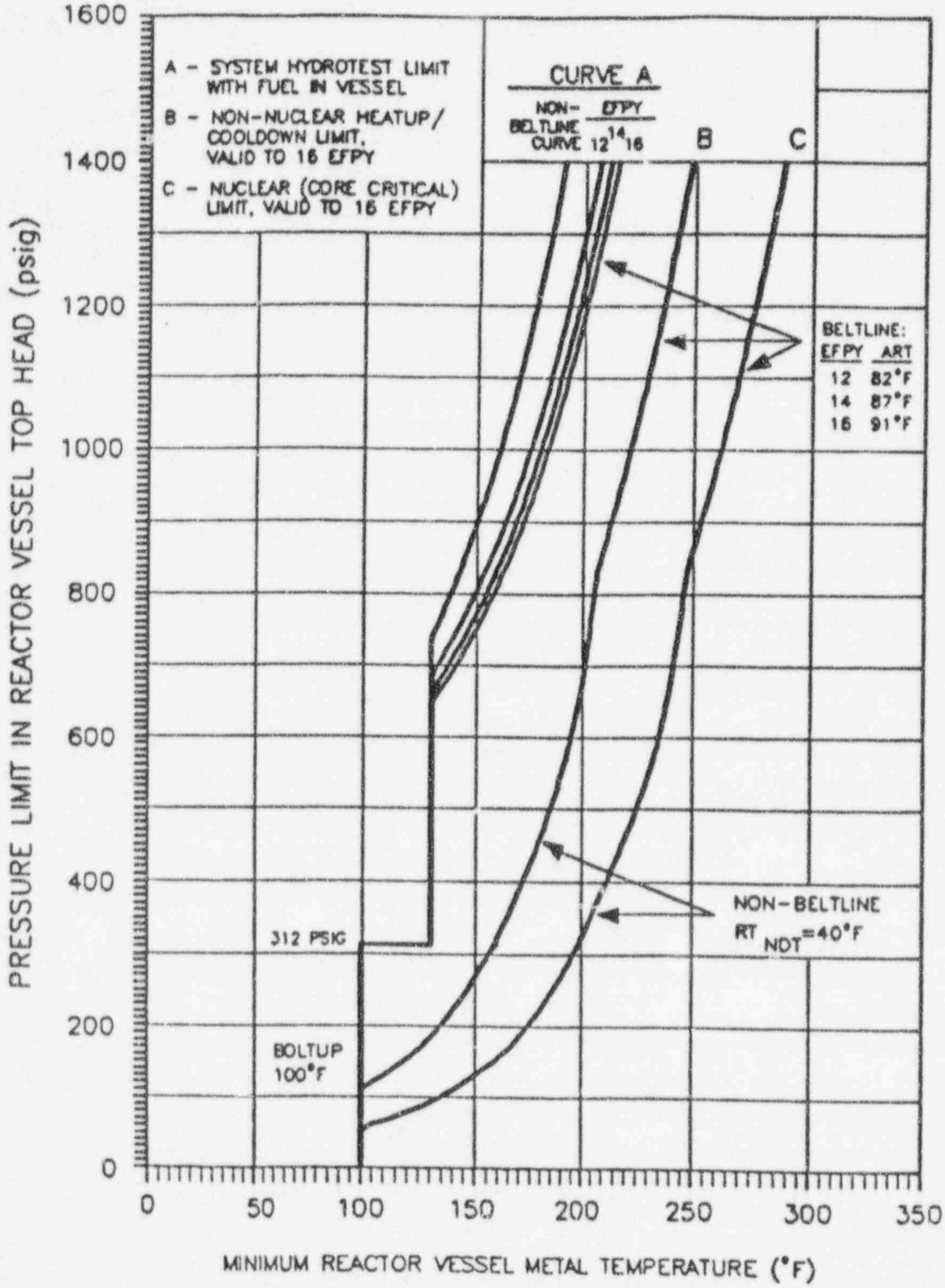
1. Restore the temperature and/or pressure to within the limits within 30 minutes, and

2. Perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system and determine that the reactor coolant system remains acceptable for continued operations, or

3. Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

FIGURE 3.6.K-1

MINIMUM REACTOR VESSEL METAL TEMPERATURE VS. REACTOR VESSEL PRESSURE



3.6 - LIMITING CONDITIONS FOR OPERATION

L. Reactor Steam Dome Pressure

The pressure in the reactor steam dome shall be ≤1005 psig.

APPLICABILITY:

OPERATIONAL MODE(s) 1[a] and 2[a]

ACTION:

With the reactor steam dome pressure >1005 psig, reduce the pressure to ≤1005 psig within 15 minutes or be in at least HOT SHUTDOWN within 12 hours.

4.6 - SURVEILLANCE REQUIREMENTS

L. Reactor Steam Dome Pressure

The reactor steam dome pressure shall be verified to be ≤1005 psig at least once per 12 hours.

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a Not applicable during anticipated transients.

3.6 - LIMITING CONDITIONS FOR OPERATION

M. Main Steam Line Isolation Valves

Two main steam line isolation valves (MSIVs) per main steam line shall be OPERABLE with closing times ≥3 seconds and ≤5 seconds.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

With one or more MSIVs inoperable, maintain at least one MSIV OPERABLE in each affected main steam line that is open and within 8 hours either:

1. Restore the inoperable valve(s) to OPERABLE status, or
2. Isolate the affected main steam line by use of a deactivated MSIV in the closed position.

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

4.6 - SURVEILLANCE REQUIREMENTS

M. Main Steam Line Isolation Valves

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.E.

3.6 - LIMITING CONDITIONS FOR OPERATION

N. Structural Integrity

The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.6.N.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, 4 and 5.

ACTION:

1. 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 limits 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. 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).

3. 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.

4.6 - SURVEILLANCE REQUIREMENTS

N. Structural Integrity

No additional Surveillance Requirements other than those required by Specification 4.0.E.

3.6 - LIMITING CONDITIONS FOR OPERATION

O. Residual Heat Removal - HOT SHUTDOWN

Two shutdown cooling mode subsystems 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 subsystem shall be capable of circulating reactor coolant[(b)] with each subsystem consisting of at least:

1. One OPERABLE RHR pump, and
2. One OPERABLE RHR heat exchanger.

APPLICABILITY:

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

ACTION:

1. With less than the above required RHR shutdown cooling mode subsystems OPERABLE, immediately initiate corrective action to return the required subsystems to OPERABLE status as soon as possible. Within 1 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 subsystem. Be in at least COLD SHUTDOWN within 24 hours.[(c)]

4.6 - SURVEILLANCE REQUIREMENTS

O. Residual Heat Removal - HOT SHUTDOWN

At least one shutdown cooling mode subsystem of the residual heat removal system, recirculation pump or alternate method shall be verified to be capable of circulating reactor coolant at least once per 12 hours.

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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.

b The RHR shutdown cooling pump may be removed from operation during hydrostatic testing.

c Whenever the two required RHR SDC 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.

3.6 - LIMITING CONDITIONS FOR OPERATION

2. With no RHR shutdown cooling mode subsystem OPERABLE, immediately initiate corrective action to return at least one subsystem to OPERABLE status as soon as possible. Within 1 hour establish reactor coolant circulation with a recirculation pump or by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

4.6 - SURVEILLANCE REQUIREMENTS

3.6 - LIMITING CONDITIONS FOR OPERATION

P. Residual Heat Removal - COLD SHUTDOWN

Two shutdown cooling mode subsystems 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 subsystem shall be capable of circulating reactor coolant[b] with each subsystem consisting of at least:

1. One OPERABLE RHR pump, and
2. One OPERABLE RHR heat exchanger.

APPLICABILITY:

OPERATIONAL MODE 4.

ACTION:

1. With less than the above required RHR shutdown cooling mode subsystems OPERABLE, within 1 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 subsystem.

4.6 - SURVEILLANCE REQUIREMENTS

P. Residual Heat Removal - COLD SHUTDOWN

At least one shutdown cooling mode subsystem of the residual heat removal system, recirculation pump or alternate method shall be verified to be capable of circulating reactor coolant at least once per 12 hours.

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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.

b The RHR shutdown cooling loop may be removed from operation during hydrostatic testing.

3.6 - LIMITING CONDITIONS FOR OPERATION

2. With no RHR shutdown cooling mode subsystem OPERABLE, immediately initiate corrective action to return at least one subsystem to OPERABLE status as soon as possible. Within 1 hour establish reactor coolant circulation with a recirculation pump or by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

4.6 - SURVEILLANCE REQUIREMENTS

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3/4.6.A Recirculation Loops

3/4.6.B Jet Pumps

3/4.6.C Recirculation Pumps

3/4.6.D Idle Recirculation Loop Startup

The reactor coolant recirculation system is designed to provide a forced coolant flow through the core to remove heat from the fuel. The reactor coolant recirculation system consists of two recirculation pump loops external to the reactor vessel. These loops provide the piping path for the driving flow of water to the reactor vessel jet pumps. The operation of the reactor coolant recirculation system is an initial condition assumed in the design basis loss-of-coolant accident (LOCA). During a LOCA caused by a recirculation loop pipe break, the intact loop is assumed to provide coolant flow during the first few seconds of the accident. The analyses assumes both loops are operating at the same flow prior to the accident. If a LOCA occurs with a flow mismatch between the two loops, the analysis conservatively assumes the pipe break is in the loop with the higher flow.

A plant specific analysis has been performed assuming only one operating recirculation loop. This analysis has demonstrated that in the event of a LOCA caused by a pipe break in the operating recirculation loop, the ECCS response will provide adequate core cooling. The transient analyses of Chapter 15 of the FSAR have also been performed for single recirculation loop operation and demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed provided the MCPR fuel cladding integrity Safety Limit is increased as noted by Specification 2.1.B. The Reactor Protection System APRM scram and control rod block setpoints are also required to be adjusted to account for the different response of the reactor and different relationships between recirculation drive flow and reactor core flow. During single loop operation for greater than 24 hours, the idle recirculation pump is electrically prohibited from starting until ready to resume two loop operation. This is done to prevent a cold water injection transient caused by an inadvertent pump startup.

Jet pump OPERABILITY is an explicit assumption in the design basis LOCA analysis. The capability of reflooding the core to two-thirds core height is dependent upon the structural integrity of the jet pumps. If a beam holding a jet pump in place fails, the jet pump suction and mixer sections could become displaced, resulting in a larger flow area through the jet pump and a lower core flooding elevation. This could adversely affect the water level in the core during the reflood phase of a LOCA as well as the assumed blowdown flow during a LOCA.

The surveillance requirements for jet pumps are designed to detect a significant degradation in jet pump performance that precedes a jet pump failure. Significant degradation is indicated if more than one of the three specified criteria confirms unacceptable deviations from established patterns or relationships. A break in a jet pump decreases the flow resistance characteristic of the external piping loop causing the recirculation pump to operate at a higher flow condition when compared to previous operation. The agreement of indicated core plate dp and core flow relationships provides assurance that the recirculation flow is not bypassing the core through inactive or broken jet pumps. The change in the flow rate of the failed jet pump produces a change in the indicated flow rate of that pump relative to the other pumps in that loop. Comparison of the data with a normal relationship or pattern provides the indication necessary to detect a failed jet pump.

The accuracy of the core flow measurement system is assumed in the derivation of the Safety Limit MINIMUM CRITICAL POWER RATIO. An analysis assuming a loss of flow indication for three jet pumps resulted in uncertainties within the values assumed for the core flow measurement system in the Safety Limit MINIMUM CRITICAL POWER RATIO calculation for both two loop operation and single loop operation. Therefore, plant operation with loss of flow indication in up to two jet pumps is acceptable as long as each jet pump is on a separate riser and no more than one calibrated double tap jet pump per loop is affected.

Recirculation pump speed mismatch limits are in compliance with the ECCS LOCA analysis design criteria. For some limited low probability events with the recirculation loop operating with large speed differences, it is possible for the LPCI loop selection logic to select the wrong loop for injection. Above 80% of RATED THERMAL POWER, the LPCI selection logic is expected to function at a speed differential of 15%. Below 80% of RATED THERMAL POWER, the loop select logic would be expected to function at a speed differential of 20%. Therefore, this specification provides a margin of 5% in pump speed differential before a problem could arise.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. The loop temperature must also be within 50°F of the reactor pressure vessel steam space coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles. Since the coolant in the bottom of the vessel is at a lower temperature than the coolant in the upper regions of the core, undue stress on the vessel would result if the temperature difference was greater than 145°F. Additionally, asymmetric speed operation of the recirculation pumps during idle loop startup induces levels of jet pump riser vibration that are higher than normal. The specific limitation of 45% of rated pump speed for the operating recirculation pump prior to the start of the idle recirculation pump ensures that the recirculation pump speed mismatch requirements are maintained.

3/4.6.E Safety Valves

3/4.6.F Relief Valves

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of safety valves are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary. The overpressure protection system must accommodate the most severe pressurization transient. Evaluations have determined that the most severe transient is the closure of all the main steam line isolation valves followed by a reactor scram on high neutron flux. The analysis results demonstrate that the design safety valve capacity is capable of maintaining reactor pressure below the ASME Code limit of 110% of the reactor pressure vessel design pressure.

The relief valve function is not assumed to operate in response to any accident, but are provided to remove the generated steam flow upon turbine stop valve closure coincident with failure of the turbine bypass system. The relief valve opening pressure settings are sufficiently low to prevent the need for safety valve actuation following such a transient.

Each of the five relief valves discharge to the suppression chamber via a dedicated relief valve discharge line. Steam remaining in the relief valve discharge line following closure can condense, creating a vacuum which may draw suppression pool water up into the discharge line. This condition is normally alleviated by the vacuum breakers; however, subsequent actuation in the presence of an elevated water leg can result in unacceptably high thrust loads on the discharge piping. To prevent this, the relief valves have been designed to ensure that each valve which closes will remain closed until the normal water level in the relief valve discharge line is restored. The opening and closing setpoints are set such that all pressure induced subsequent actuation are limited to the two lowest set valves. These two valves are equipped with additional logic which functions in conjunction with the setpoints to inhibit valve reopening during the elevated water leg duration time following each closure.

Each safety/relief valve is equipped with diverse position indicators which monitor the tailpipe acoustic vibration and temperature. Either of these provide sufficient indication of safety/relief valve position for normal operation.

<u>3/4.6.G</u> <u>Leakage Detection Systems</u>

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. Limits on leakage from the reactor coolant pressure boundary are required so that appropriate action can be taken before the integrity of the reactor coolant pressure boundary is impaired. Leakage detection systems for the reactor coolant system are provided to alert the operators when leakage rates above the normal background levels are detected and also to supply quantitative measurement of leakage rates. Leakage from the reactor coolant pressure boundary inside the drywell is detected by at least one or two independently monitored variables, such as sump level changes and drywell atmosphere radioactivity levels. The means of quantifying leakage in the drywell is the drywell floor drain sump pumps. With the drywell floor drain sump pump system inoperable, no other form of monitoring can provide the equivalent information. However, primary containment atmosphere sampling for radioactivity can provide indication of changes in leakage rates.

3/4.6.H Operational Leakage

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shutdown to allow further investigation and corrective action.

An UNIDENTIFIED LEAKAGE increase of more than 2 gpm within a 24 hour period is an indication of a potential flaw in the reactor coolant pressure boundary and must be quickly evaluated. Although the increase does not necessarily violate the absolute UNIDENTIFIED LEAKAGE limit, IGSCC susceptible components must be determined not to be the source of the leakage within the required completion time.

3/4.6.I Chemistry

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low, thus the 0.2 ppm limit on chlorides is permitted during POWER OPERATION.

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity meter inoperable, additional samples must be analyzed to ensure that the chlorides are not exceeding the limits.

Action 1 permits temporary operation with chemistry limits outside of the limits required in OPERATIONAL MODE 1 without requiring Commission notification. The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

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3/4.6.J Specific Activity

The limitations on the specific activity of the primary coolant ensure that the 2 hour thyroid and whole body doses resulting from a main steam line failure outside the containment during steady state operation will not exceed small fractions of the dose guidelines of 10 CFR 100. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131, but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analysis following power changes may be permissible if justified by the data obtained.

Closing the main steam line isolation valves prevents the release of activity to the environs should a steam line rupture occur outside containment. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

3/4.6.K Pressure/Temperature Limits

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9.1.1.1 of the UFSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

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

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

The pressure-temperature limit lines shown in Figure 3.6.K-1, for operating conditions; Inservice Hydrostatic Testing (curve A), Non-Nuclear Heatup/Cooldown (curve B), and Core Critical Operation (curve C). The curves have been established to be in conformance with Appendix G to 10 CFR Part 50 and Regulatory Guide 1.99 Revision 2, and take into account the change in reference nil-ductility transition temperature (RT_{NDT}) as a result of neutron embrittlement. The adjusted reference temperature (ART) of the limiting vessel material is used to account for irradiation effects.

Three vessel regions are considered for the development of the pressure-temperature curves: 1) the core beltline region; 2) the non-beltline region (other than the closure flange region); and 3) the closure flange region. The beltline region is defined as that region of the reactor vessel that directly surrounds the effective height of the reactor core and is subject to an RT_{NDT} adjustment to account for radiation embrittlement. The non-beltline and closure flange regions receive insufficient fluence to necessitate an RT_{NDT} adjustment. These regions contain components which include; the reactor vessel nozzles, closure flanges, top and bottom head plates, control rod drive penetrations, and shell plates that do not directly surround the reactor core. Although the closure flange region is a non-beltline region, it is treated separately for the development of the pressure-temperature curves to address 10CFR Part 50 Appendix G requirements.

In evaluating the adequacy of the steel which comprises the reactor vessel, it is necessary that the following be established: 1) the RT_{NDT} for all vessel and adjoining materials; 2) the relationship between RT_{NDT} and integrated neutron flux (fluence, at energies greater than one Mev); and 3) the fluence at the location of a postulated flaw.

Boltup Temperature

The initial RT_{NDT} of the main closure flanges, the shell and head materials connecting to these flanges, and connecting welds is 10°F; however, the vertical electroslag welds which terminate immediately below the vessel flange have an RT_{NDT} of 40°F. Therefore, the minimum allowable boltup temperature is established as 100°F (RT_{NDT} + 60°F) which includes a 60°F conservatism required by the original ASME Code of construction.

Curve A - Hydrotesting

As indicated in curve A of Figure 3.6.K-1 for system hydrotesting, the minimum metal temperature of the reactor vessel shell is 100°F for reactor pressures less than 312 psig. This 100°F minimum boltup temperature is based on a RT_{NDT} of 40°F for the electroslag weld immediately below the vessel flange and a 60°F conservatism required by the original ASME Code of construction. At reactor pressures greater than 312 psig, the minimum vessel metal temperature is established as 130°F. The 130°F minimum temperature is based on a closure flange region RT_{NDT} of 40°F and a 90°F conservatism required by 10CFR Part 50 Appendix G

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for pressure in excess of 20% of the preservice hydrostatic test pressure (1563 psig). At approximately 650 psig the effects of pressurization are more limiting than the boltup stresses at the closure flange region, hence a family of non-linear curves intersect the 130°F vertical line. Beltline as well as non-beltline curves have been provided to allow separate monitoring of the two regions. Beltline curves as a function of vessel exposure for 12, 14 and 16 effective full power years (EFPY) are presented to allow the use of the appropriate curve up to 16 EFPY of operation.

A typical sequence involved in pressure testing is a heatup to the required temperature and then pressurization to the required pressure for the inspection. During the heatup, at 100°F/hour or less, Curve B is the governing curve. Since the vessel is not pressurized during the heatup, Curves A and B are the same. When temperatures are stabilized to within 20°F/hour rates, at temperatures above those required by curve A, pressurization begins, at which point Curve A is the governing curve. During the inspection period with the vessel at the required pressure, temperature changes are limited to 20°F/hour.

<u>Curve B - Non-Nuclear Heatup/Cooldown</u>

Curve B of Figure 3.6.K-1 applies during heatups with non-nuclear heat (e.g., recirculation pump heat) and during cooldowns when the reactor is not critical (e.g., following a scram). The curve provides the minimum reactor vessel metal temperatures based on the most limiting vessel stress. As indicated by the vertical 100°F line, the boltup stresses at the closure flange region are most limiting for reactor pressures below approximately 110 psig. For reactor pressures greater than approximately 110 psig, pressurization and thermal stresses become more limiting than the boltup stresses, which is reflected by the nonlinear portion of curve B. The non-linear portion of the curve is dependent on non-beltline and beltline regions, with the beltline region temperature limits having been adjusted to account for vessel irradiation (up to a vessel exposure of 16 EFPY). The non-beltline region is limiting between approximately 110 psig and 830 psig. Above approximately 803 psig, the beltline region becomes limiting.

<u>Curve C - Core Critical Operation</u>

Curve C, the core critical operation curve shown in Figure 3.6.K-1, is generated in accordance with 10CFR Part 50 Appendix G which requires core critical pressure-temperature limits to be 40°F above any curve A or B limits. Since curve B is more limiting, (curve C is curve B plus 40°F.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTI E185-73 and 10CFR Part 50, Appendix H, irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel in the core area. The irradiated specimens can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figure 3.6.J-1 shall be adjusted, as required, on the basis of the specimen data and recommendations of Regulatory Guide 1.99, Revision 2.

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3/4.6.L Reactor Steam Dome

The reactor steam dome pressure is an assumed initial condition of Design Basis Accidents and transients and is also an assumed value in the determination of compliance with reactor pressure vessel overpressure protection criteria. The reactor steam dome pressure of ≤1005 psig is an initial condition of the vessel overpressure protection analysis. This analysis assumes an initial maximum reactor steam dome pressure and evaluates the response of the pressure relief system, primarily the safety valves, during the limiting pressurization transient. The determination of compliance with the overpressure criteria is dependent on the initial reactor steam dome pressure; therefore, the limit on this pressure ensures that the assumptions of the overpressure protection analysis are conserved.

3/4.6.M Main Steam Line Isolation Valves

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

3/4.6.N Structural Integrity

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

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

3/4.6.O Residual Heat Removal - HOT SHUTDOWN

3/4.6.P Residual Heat Removal - COLD SHUTDOWN

Irradiated fuel in the reactor pressure vessel generates decay heat during normal and abnormal shutdown conditions, potentially resulting in an increase in the temperature of the reactor coolant. This decay heat is required to be removed such that the reactor coolant temperature can be reduced in preparation for performing refueling, maintenance operations or for maintaining the reactor in cold shutdown conditions. Systems capable of removing decay heat are therefore required to perform these functions.

A single shutdown cooling mode subsystem provides sufficient heat removal capability for removing core decay heat and mixing to assure accurate temperature indication, however, single failure considerations require that two subsystems be OPERABLE or that alternate methods capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation. An OPERABLE RHR shutdown cooling subsystem consists of one OPERABLE RHR pump, one heat exchanger, and the associtated piping and valves. The two subsystems have a common suction source and are allowed to have a common heat exchange, and common discharge piping. Therefore, to meet the Limiting Condition for Operation, both pumps in one loop or one pump in each of the two loops must be OPERABLE. Since the piping and heat exchangers are passive components that are assumed not to fail, they are allowed to be common to both subsystems (the ability to take credit for a common heat exchanger and discharge piping only applies to the SDC mode of RHR).