50-237/249
50-254/265

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

September 21, 1995

Mr. D. L. Farrar
Manager, Nuclear Regulatory Services
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 500
Downers Grove, Illinois 60515

SUBJECT: ISSUANCE OF AMENDMENTS (TAC NOS. M90401, M90402, M90403 AND M90404)

Dear Mr. Farrar:

The Commission has issued the enclosed Amendment No. 140 to Facility Operating License No. DPR-19 and Amendment No. 134 to Facility Operating License No. DPR-25 for the Dresden Nuclear Power Station, Units 2 and 3, respectively; and Amendment No. 162 to Facility Operating License No. DPR-29 and Amendment No. 158 to Facility Operating License No. DPR-30 for the Quad Cities Nuclear Power Station, Units 1 and 2, respectively. The amendments are in response to your application dated September 17, 1993, as supplemented by letter dated June 30, 1995.

As a result of findings by a Diagnostic Evaluation Team inspection performed by the NRC staff at the Dresden Nuclear Power Station in 1987, Commonwealth Edison Company (ComEd, the licensee) made a decision that both the Dresden Nuclear Power Station and sister site Quad Cities Nuclear Power Station, needed attention focused on the existing custom Technical Specifications (TS) being used at both sites.

The licensee made the decision to initiate a Technical Specification Upgrade Program (TSUP) for both Dresden and Quad Cities. The licensee evaluated the current TS for both Dresden and Quad Cities against the Standard Technical Specifications (STS) contained in NUREG-0123, "Standard Technical Specification General Electric Plants BWR/4." The licensee's evaluation identified numerous potential improvements such as clarifying requirements, changing the TS to make them more understandable and to eliminate interpretation, and deleting requirements that are no longer considered current with industry practice. As a result of the evaluation, ComEd has elected to upgrade both the Dresden and Quad Cities TS to the STS contained in NUREG-0123.

The TSUP for Dresden and Quad Cities is not a complete adoption of the STS. The TSUP focuses on (1) integrating additional information such as equipment operability requirements during shutdown conditions, (2) clarifying requirements such as limiting conditions for operations and action statements utilizing STS terminology, (3) deleting superseded requirements and modifications to the TS based on the licensee's responses to Generic Letters (GL), and (4) relocating specific items to more appropriate TS locations.

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The application dated September 17, 1993, as supplemented June 30, 1995, contains the proposed upgrade of Section 3/4.6 (Primary System Boundary) of the Dresden and Quad Cities TS.

The review guidance to be used by the NRC staff in the review of the TSUP is described in Section 2.0 of the enclosed Safety Evaluation (SE). The staff reviewed the proposed changes and evaluated all deviations and changes between the proposed TS, the STS, and the current TS.

Based on discussions between ComEd and the staff, it has been mutually agreed upon that the NRC will review the sections of TSUP as they are submitted and provide ComEd an amendment for each submittal. Once all of the TSUP sections have been reviewed and the amendments issued, it is our understanding that ComEd will make one final submittal addressing any changes that may be required as a result of problems uncovered during the course of this effort. Upon receipt and review of this final submittal, the staff will issue a final amendment which addresses any remaining open items and any changes or corrections to the previous amendments. The applicable TSUP TS will be issued with each amendment and will become effective no later than December 31, 1995, for Dresden and June 30, 1996, for Quad Cities.

The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

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

Docket Nos. 50-237, 50-249, 50-254, 50-265

Enclosures:
1. Amendment No. 140 to DPR-19
2. Amendment No. 134 to DPR-25
3. Amendment No. 162 to DPR-29
4. Amendment No. 158 to DPR-30
5. Safety Evaluation

cc w/encls: see next page

D. L. Farrar
Commonwealth Edison Company

Dresden Nuclear Power Station
Unit Nos. 2 and 3
Quad Cities Nuclear Power Station
Unit Nos. 1 and 2

cc:

Michael I. Miller, Esquire
Sidley and Austin
One First National Plaza
Chicago, Illinois 60603

Mr. Thomas P. Joyce
Site Vice President
Dresden Nuclear Power Station
6500 North Dresden Road
Morris, Illinois 60450-9765

Mr. J. Heffley
Station Manager
Dresden Nuclear Power Station
6500 North Dresden Road
Morris, Illinois 60450-9765

U.S. Nuclear Regulatory Commission
Resident Inspectors Office
Dresden Station
6500 North Dresden Road
Morris, Illinois 60450-9766

Richard J. Singer
Manager - Nuclear
MidAmerican Energy Company
907 Walnut Street
P.O. Box 657
Des Moines, Iowa 50303

Brent E. Gale, Esq.
Vice President - Law and
Regulatory Affairs
One RiverCenter Place
106 East Second Street
P.O. Box 4350
Davenport, Iowa 52808

Chairman
Rock Island County Board
of Supervisors
1504 3rd Avenue
Rock Island County Office Bldg.
Rock Island, Illinois 61201

Regional Administrator
U.S. NRC, Region III
801 Warrenville Road
Lisle, Illinois 60532-4351

Illinois Department of Nuclear Safety
Office of Nuclear Facility Safety
1035 Outer Park Drive
Springfield, Illinois 62704

Chairman
Grundy County Board
Administration Building
1320 Union Street
Morris, Illinois 60450

Mr. Stephen E. Shelton
Vice President
Iowa-Illinois Gas and
Electric Company
P. O. Box 4350
Davenport, Iowa 52808

Mr. L. William Pearce
Station Manager
Quad Cities Nuclear Power Station
22710 206th Avenue North
Cordova, Illinois 61242

U.S. Nuclear Regulatory Commission
Quad Cities Resident Inspectors Office
22712 206th Avenue North
Cordova, Illinois 61242

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:

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(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

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.

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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.

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.

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.

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

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

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-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

ACTION:

With any of the above limits exceeded,

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

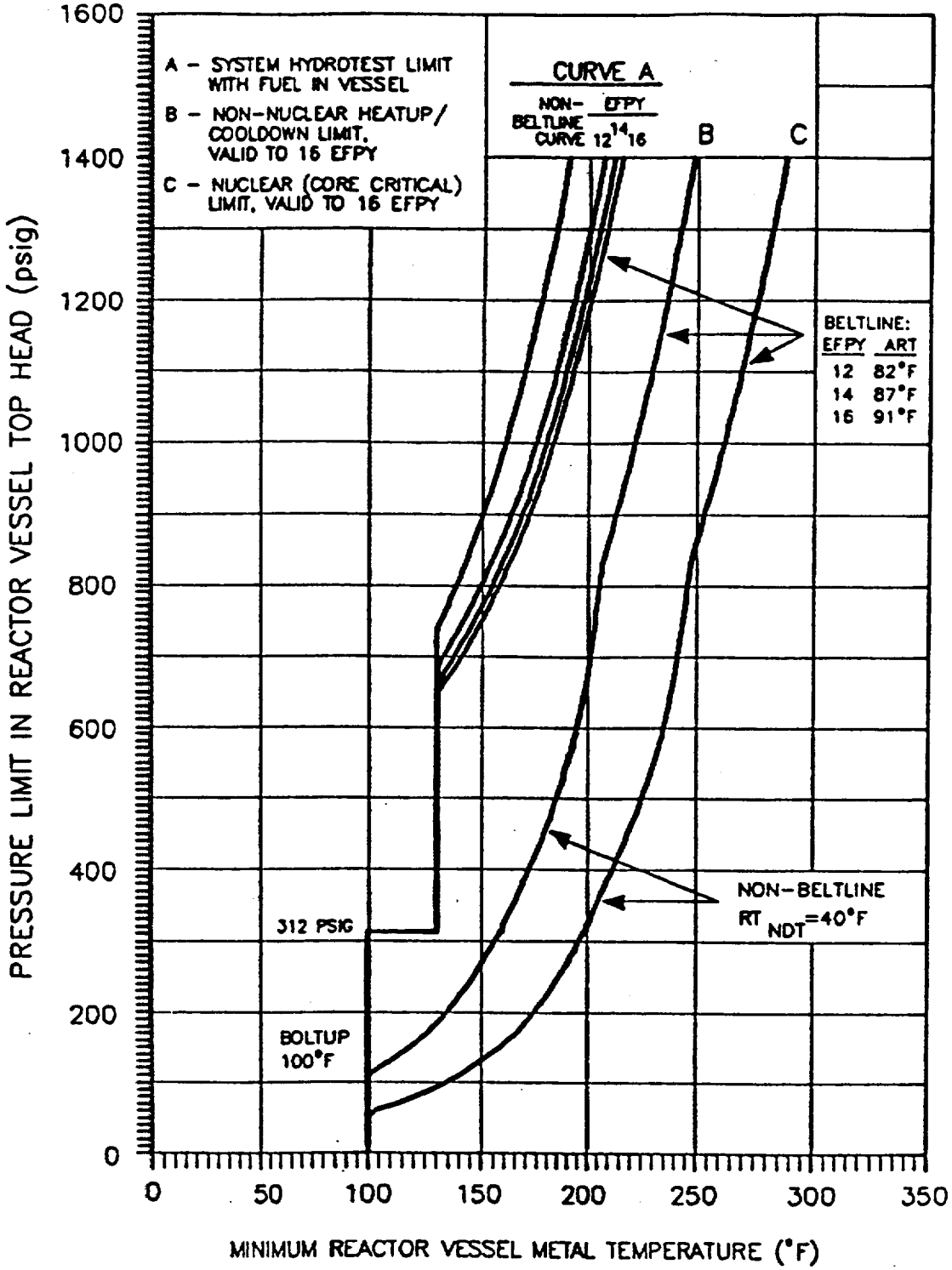
2. Perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system and determine that the reactor coolant system remains acceptable for continued operations, or

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

4.6 - SURVEILLANCE REQUIREMENTS

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

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.

4.6 - SURVEILLANCE REQUIREMENTS

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

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.

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

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.

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

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.

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. 1[illegible] 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
3.6/4.6-5	3.6/4.6-4	3/4.6-5
3.6/4.6-6	3.6/4.6-4a	3/4.6-6
3.6/4.6-7	3.6/4.6-4b	3/4.6-7
3.6/4.6-8	3.6/4.6-5	3/4.6-8
3.6/4.6-9	3.6/4.6-5a	3/4.6-9
3.6/4.6-10	3.6/4.6-5b	3/4.6-10
3.6/4.6-11	3.6/4.6-5b(i)	3/4.6-11
3.6/4.6-12	3.6/4.6-5c	3/4.6-12
3.6/4.6-12a	3.6/4.6-5d	3/4.6-13
3.6/4.6-13	3.6/4.6-5e	3/4.6-14
3.6/4.6-14	3.6/4.6-5f	3/4.6-15
3.6/4.6-15	3.6/4.6-5g	3/4.6-16
3.6/4.6-15a	3.6/4.6-5h	3/4.6-17
3.6/4.6-15b	3.6/4.6-6	3/4.6-18
3.6/4.6-15c	3.6/4.6-7	3/4.6-19
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
3.6/4.6-23	3.6/4.6-14	B 3/4.6-1
3.6/4.6-24	3.6/4.6-14a	B 3/4.6-2
3.6/4.6-25	Figure 3.6-1	B 3/4.6-3
3.6/4.6-25a	3.6/4.6-16	B 3/4.6-4
3.6/4.6-26	3.6/4.6-17	B 3/4.6-5
3.6/4.6-27	3.6/4.6-18	B 3/4.6-6
3.6/4.6-28	3.6/4.6-19	B 3/4.6-7
3.6/4.6-29	3.6/4.6-20	B 3/4.6-8
3.6/4.6-30	3.6/4.6-21	B 3/4.6-9
3.6/4.6-31	3.6/4.6-21A	---
3.6/4.6-32	3.6/4.6-22	---
3.6/4.6-33	Figure 4.6-1	---
3.6/4.6-34	Figure 4.6-2	---
3.6/4.6-35	---	---
3.6/4.6-36	---	---
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.

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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 ≤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 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.

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

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-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

ACTION:

With any of the above limits exceeded,

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

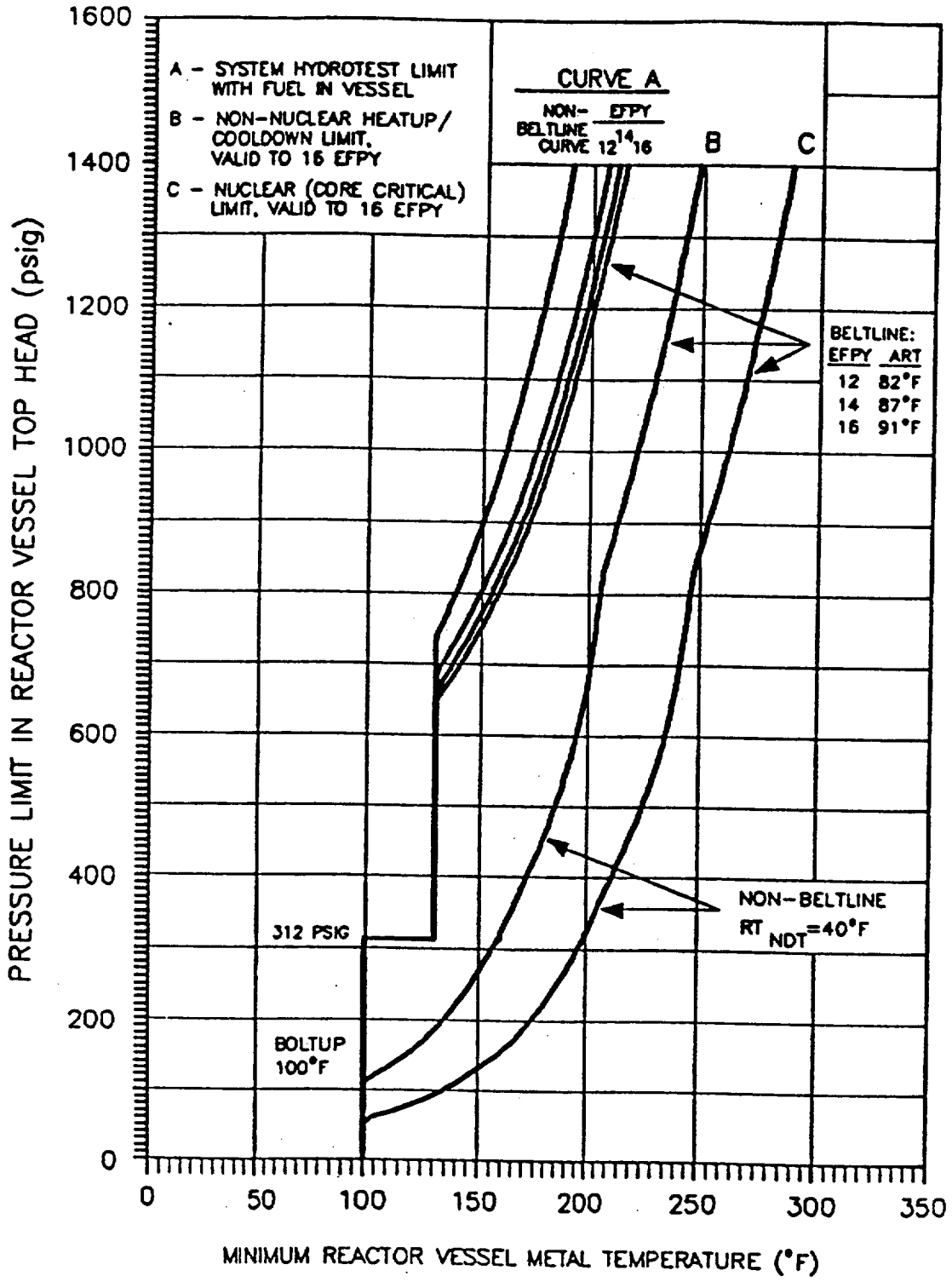
2. Perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system and determine that the reactor coolant system remains acceptable for continued operations, or

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

4.6 - SURVEILLANCE REQUIREMENTS

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. 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.

---

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

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

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.

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 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

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.

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 exchanger 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).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 140 TO FACILITY OPERATING LICENSE NO. DPR-19,

AMENDMENT NO. 134 TO FACILITY OPERATING LICENSE NO. DPR-25,

AMENDMENT NO. 162 TO FACILITY OPERATING LICENSE NO. DPR-29,

AND AMENDMENT NO. 158 TO FACILITY OPERATING LICENSE NO. DPR-30

COMMONWEALTH EDISON COMPANY

AND

MIDAMERICAN ENERGY COMPANY

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-237, 50-249, 50-254 AND 50-265

1.0 INTRODUCTION

By letter dated September 17, 1993, as supplemented by letter dated June 30, 1995, Commonwealth Edison Company (ComEd, the licensee) submitted an amendment requesting to upgrade sections of the Dresden Nuclear Power Station, Units 2 and 3, and the Quad Cities Nuclear Power Station, Units 1 and 2, Technical Specifications (TS). The changes have been requested as part of their Technical Specification Upgrade Program (TSUP).

As a result of findings by a Diagnostic Evaluation Team inspection performed by the NRC staff at the Dresden Nuclear Power Station in 1987, ComEd made a decision that both the Dresden Nuclear Power Station and sister site Quad Cities Nuclear Power Station, needed attention focused on the existing custom TS used at the sites.

The licensee made the decision to initiate a TSUP for both Dresden and Quad Cities. The licensee evaluated the current TS for both stations against the Standard Technical Specifications (STS), contained in NUREG-0123, "Standard Technical Specifications General Electric Plants BWR/4." Both Dresden and Quad Cities are BWR-3 designs and are nearly identical plants. The licensee's evaluation identified numerous potential improvements such as clarifying requirements, changing the TS to make them more understandable and to eliminate the need for interpretation, and deleting requirements that are no longer considered current with industry practice. As a result of the evaluation, ComEd elected to upgrade both the Dresden and Quad Cities TS to the STS contained in NUREG-0123.

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The TSUP for Dresden and Quad Cities is not a complete adoption of the STS. The TSUP focuses on (1) integrating additional information such as equipment operability requirements during shutdown conditions, (2) clarifying requirements such as limiting conditions for operations and action statements utilizing STS terminology, (3) deleting superseded requirements and modifications to the TS based on the licensee's responses to Generic Letters (GL), and (4) relocating specific items to more appropriate TS locations or to licensee controlled documents.

The application dated September 17, 1993, as supplemented June 30, 1995, proposed to upgrade only those sections of the TS to be included in TSUP Section 3/4.6 (Primary System Boundary) of the Dresden and Quad Cities TS.

The staff reviewed the proposed changes and evaluated all deviations and changes between the proposed TS, the STS, and the current TS. In no case did the licensee propose a change in the TS that would result in the relaxation of the current design requirements as stated in the Updated Final Safety Analysis Reports (UFSAR) for Dresden or Quad Cities.

In response to the staff's recommendations, the licensee submitted identical TS for Quad Cities and Dresden except for plant-specific equipment and design differences. Technical differences between the units are identified as appropriate in the proposed amendment.

2.0 EVALUATION

Review Guidelines - The licensees' purpose for the TSUP was to reformat the existing Dresden and Quad Cities TS into the easier to use STS format. Plant specific data, values, parameters, and equipment specific operational requirements contained in the current TS for Dresden and Quad Cities were retained by the licensee in the TSUP.

The STS contained in NUREG-0123 were developed by the NRC and industry because of the shortcomings associated with the custom TS which were issued to plants licensed in early 1970's (i.e., Dresden (1971) and Quad Cities (1972)). The STS developed by the NRC and industry provided an adequate level of protection for plant operation by assuring required systems are operable and have been proven to be able to perform their intended functions. The limiting conditions for operation (LCO), the allowed out-of-service times, and the required surveillance frequencies were developed based on industry operating experience, equipment performance, and probabilistic risk assessment analysis during the 1970's. The STS were used as the licensing basis for plants licensed starting in the late 1970's.

For the most part, ComEd's adoption of the STS resulted in more restrictive LCOs and surveillance requirements (SR). In some cases, however, the STS provides relief from the Dresden and Quad Cities current TS requirements. In all these cases, the adoption of the STS requirements for LCOs or SRs do not change the current design requirements of either plant as described in the each plant's UFSAR. In addition, the success criteria for the availability

and operability of all required systems contained in the current TS are maintained by the adoption of the STS requirements in the proposed TSUP TS.

In addition to adopting the STS guidelines and requirements in the TSUP, ComEd has also evaluated GLs concerning line item improvements for TS. These GLs were factored into TSUP to make the proposed TS in the TSUP reflect industry lessons learned in the 1980's and early 1990's.

Deviations between the proposed specifications, the STS, and the current TS were reviewed by the staff to determine if they were due to plant specific features or if they posed a technical deviation from the STS guidelines. Plant specific data, values, parameters, and equipment specific operational requirements contained in the current TS for Dresden and Quad Cities were retained by the licensee in the upgraded TS.

Administrative Changes - Non-technical, administrative changes were intended to incorporate human factor principles into the form and structure of the STS so that they would be easier for plant operation's personnel to use. These changes are editorial in nature or involve the reorganization or reformatting of requirements without affecting technical content of the current TS or operational requirements. Every section of the proposed TS reflects this type of change.

More Restrictive Requirements - The proposed TSUP TS include certain more restrictive requirements than are contained in the existing TS. Examples of more restrictive requirements include the following: placing an LCO on plant equipment which is not required by the present TS to be operable; adding more restrictive requirements to restore inoperable equipment; and adding more restrictive SR.

Less Restrictive Requirements - The licensee provided a justification for less restrictive requirements on a case-by-case basis as discussed in this Safety Evaluation (SE). When requirements have been shown to provide little or no safety benefit, their removal from the TS may be appropriate. In most cases, these relaxations had previously been granted to individual plants on a plant-specific basis as the result of (a) generic NRC actions, and (b) new NRC staff positions that have evolved from technological advancements and operating experience.

The Dresden and Quad Cities plant designs were reviewed to determine if the specific design basis was consistent with the STS contained in NUREG-0123. All changes to the current TS and deviations between the licensees' proposed TS and the STS were reviewed by the staff for acceptability to determine if adequate justification was provided (i.e., plant specific features, retention of existing operating values, etc.).

Deviations the staff finds acceptable include: (1) adding clarifying statements, (2) incorporating changes based on GLs, (3) reformatting multiple steps included under STS action statements into single steps with unique identifiers, (4) retaining plant specific steps, parameters, or values,

(5) moving action statements within a TS, (6) moving action statements from an existing TS to form a new TS section, and (7) omitting the inclusion of STS steps that are not in existing TS.

Relocation of Technical Specifications - The proposed TS may include the relocation of some requirements from the TS to licensee-controlled documents. Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to state TS to be included as part of the license. The Commission's regulatory requirements related to the content of TS are set forth in 10 CFR 50.36. That regulation requires that the TS include items in five specific categories, including (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. However, the regulation does not specify the particular requirements to be included in a plant's TS.

The Commission has provided guidance for the contents of TS in its "Final Policy Statement on Technical Specification Improvements for Nuclear Power Reactors," 58 Federal Register 39132 (July 22, 1993), in which the Commission indicated that compliance with the Final Policy Statement satisfies Section 182a of the Energy Reorganization Act. The Final Policy Statement identified four criteria to be used in determining whether a particular matter is required to be included in the TS, as follows: (1) installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary; (2) a process variable, design feature, or operating restriction that is an initial condition of a Design Basis Accident or Transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; (3) a structure, system, or component that is part of a primary success path and which functions or actuates to mitigate a Design Basis Accident of Transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; (4) a structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety. As a result, existing TS requirements which fall within or satisfy any of the criteria in the Final Policy Statement must be retained in the TS, while those TS requirements which do not fall within or satisfy these criteria may be relocated to other, licensee-controlled documents. The Commission recently amended 10 CFR 50.36 to codify and incorporate these four criteria (60 FR 36953). The change to 10 CFR 50.36 is effective as of August 18, 1995.

The following sections provide the staff's evaluations of the specific proposed TS changes.

3.0 EVALUATION OF TSUP PROPOSED TS SECTION 3/4.6, PRIMARY SYSTEM BOUNDARY

The following sections provide the staff's evaluation of the TS changes reflected in proposed TSUP TS Section 3/4.6. Proposed TS Section 3/4.6 results from the reordering of current TS Section 3/4.6 based on STS format

and nomenclature. All deviations between the current and proposed TS, and between the proposed TS and STS guidelines, are discussed below.

3.1 Section 3/4.6.A, Recirculation Loops

Proposed TS 3/4.6.A, "Recirculation Loops" incorporates the guidance of the STS Section 3/4.4.1.1 and requirements from Section 3/4.6.H of the current TS for both stations. The proposed TS has been reformatted adopting the STS guidelines. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR.

3.1.1 Actions

Current TS 3.6.H.3 discusses the requirements when only one recirculation loop is in operation. The current TS require a specific percentage reduction for the minimum critical power ratio (MCPR), average power range monitor (APRM), and average planar linear heat generation rate (APLHGR) safety limits under this circumstance. Proposed action 1.c deviates from the current TS by referring to Section 3.2 of the proposed TS for details regarding the required reduction in these limits. This is an administrative change to relocate requirements to a separate section of the TS and is acceptable.

Current Dresden TS 3.6.H.3.g discusses the requirements for one Automatic Pressure Relief Subsystem relief valve out-of-service. This requirement will be relocated to the proposed TS Section 3/4.5. This change is administrative and is acceptable.

Proposed TS 3.6.A action 1.d specifies that the minimum average planar linear heat generation rate (MAPLHGR) limits shall be reduced by the appropriate factors as specified in the core operating limits report. This is a new requirement for Quad Cites not included in the current TS. Therefore, the proposed TS is more conservative and is acceptable. This requirement is in the current Dresden TS and therefore, there is no change for Dresden.

3.1.2 Surveillance Requirements

Proposed TS 4.6.A is a new requirement to demonstrate operability of the pump motor generator set overspeed setpoints. The proposed TS is an enhancement of current TS and is acceptable.

3.1.3 Conclusion

The staff finds that the proposed TS Section 3/4.6.A, "Recirculation Loops" has been reformatted adopting the STS guidelines. The staff has reviewed the proposed TS against the STS and current TS requirements and finds that there are no significant deviations from STS and the proposed TS do not relax any existing TS requirements. Therefore, the staff finds proposed TS Section 3/4.6.A acceptable.

3.2 Section 3/4.6.B, Jet Pumps

Proposed TS 3/4.6.B, "Jet Pumps" incorporates the guidance of the STS Section 3/4.4.1.2 and requirements from Section 3/4.6.G of the current TS for both stations. The proposed TS has been reformatted adopting the STS guidelines. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR.

3.2.1 Actions

Proposed action 1 requires that, with one or more jet pumps inoperable for other than inoperable flow indication, the unit must be in hot shutdown within 12 hours. The current TS deviate from the proposed by not specifying the exception for inoperable flow indication. The current TS do not indicate whether inoperable flow indication renders the jet pump inoperable. Current operating practice has been to not enter the action statement (cold shutdown within 24 hours) if flow indication is inoperable. The proposed TS clarify the definition of an operable jet pump. Therefore, the proposed TS is a clarification of current TS and is therefore acceptable.

The proposed TS also deviates from the current TS in that the proposed requirement to be in hot shutdown within 12 hours is a change from the current requirement to be in cold shutdown within 24 hours. Because the proposed applicability is only modes 1 and 2, going to hot shutdown brings the plant to a condition in which jet pumps are not required to be operable. In addition, the proposed requirement ensures that the reactor is placed in a safe condition in a more expeditious time frame (12 hours versus 24 hours). There is no improvement in safety gained by placing the reactor in cold shutdown and doing so may result in unnecessary transients. Therefore, this change is acceptable.

Current Dresden TS 3.6.G.2 requires verification of flow indication from each of the jet pumps prior to initiation of reactor startup. However, current Dresden TS 3.6.G.4 allows continued operation with one inoperable flow indication for the jet pumps. Current Quad Cities TS 3.6.G.2 requires verification of flow indication from all but one jet pump prior to startup. Current Quad Cities TS 3.6.G.3 allows continued operation with two inoperable flow indicators. The action proposed in the September 17, 1993, submittal would allow an indefinite period of operation with two inoperable jet pump flow indicators. This proposed action is consistent with current Quad Cities TS but would be a relaxation of the current Dresden TS which allow a 12 hour allowed outage time (AOT) to restore one inoperable jet pump flow indicator. This is an open item for Dresden Station, contingent upon its review and approval in the clean-up amendment.

Current TS 3.6.G.3 provides the definition of indicated core flow. Current Dresden TS refers to station procedures for adjusting the indicated flow during single loop operation (SLO). Current Quad Cities TS provide corrective actions for jet pumps with inoperable flow indication. Current TS 3.6.G.3 is not retained in the proposed TS. The current TS contains details related to

system design which are inappropriate for inclusion within the TS and are more appropriately controlled by plant procedures. The staff has determined that these details are not required to be in the TS under 10 CFR 50.36 or Section 182a of the Atomic Energy Act. Further, they do not fall within any of the four criteria discussed in Section 2.0 above. Therefore, the deletion of this TS is acceptable.

3.2.2 Surveillance Requirements

Proposed TS 4.6.B.1 deviates from the current TS by specifying that the surveillances must be performed at greater than 25% rated thermal power. The current TS require that the surveillances be performed whenever there is recirculation flow with the reactor in the startup/hot standby or run modes. The proposed TS delays performance of the surveillances until reactor power has reached 25% because during low flow conditions, jet pump flow measurement is unachievable. The proposed SR is adequate to detect significant degradation in jet pump performance that precedes a jet pump failure and is acceptable. The proposed requirements are based on STS guidelines but deviate from the STS guidelines by requiring that the jet pump surveillances be performed after exceeding 25 percent power while the STS require the surveillance prior to exceeding 25 percent of rated thermal power. This deviation from STS requirements is acceptable based on the inability to accurately measure jet pump flow during low flow conditions.

Proposed TS 4.6.B.2 is a modification of current TS 4.6.G.2. The current TS requires that the differential pressure of any jet pump in the idle loop shall not vary by more than 10 percent from estimated patterns. The proposed TS require that the recirculation pump flow, total core flow, and indicated flow of any individual jet pump doesn't differ by more than 10 percent from estimated patterns. The proposed TS is more restrictive than the current requirements and is acceptable.

Proposed TS 4.6.B.1.d and 4.6.B.2.d are new TS which state that the provisions of proposed TS 4.0.D are not applicable if the SR is performed within 24 hours after exceeding 25 percent of rated thermal power. Proposed TS 4.0.D requires that entry into an operational mode may not be made unless the associated LCOs have been performed. Because the proposed TS doesn't require that the surveillances be performed until after reaching 25% of rated thermal power, the allowance for a minimum period of time (24 hours) is necessary in order to satisfy the SR. Therefore, this change is acceptable.

Proposed TS 4.6.B.1.c is a new requirement for Dresden and is consistent with current Quad Cities TS 4.6.G.1.c. The proposed TS is an enhancement of current Dresden TS and is acceptable.

Current TS 3.6.G.2 was not retained in the proposed TS. This current TS requires verification of flow indication from each of the jet pumps prior to initiation of startup. The intention of this TS is to ensure jet pump flow indication upon startup of the reactor. The requirement to verify flow indication is unnecessary as this is verified during the surveillances to

determine jet pump flow. Proposed TS 4.6.B.1 and 4.6.B.2 require measurement of jet pump flow within 24 hours of exceeding 25 percent of rated thermal power. Although the current TS require this verification prior to initiation of startup and proposed TS 4.6.B.1 and 4.6.B.2 require measurement after achieving 25% power, as discussed above, during cold shutdown or low flow conditions, flow indication is unachievable. The proposed surveillances provide an equivalent level of jet pump indication during startup when the conditions are appropriate for performing the surveillance. Therefore, the deletion of current TS 3.6.G.2 is acceptable.

Current TS 4.6.G.3 discusses how the baseline test data is acquired and has been deleted in the proposed TS. Current TS 4.6.G.3 contains details related to the methods for performing surveillances which are inappropriate for inclusion within the TS and are more appropriately controlled by plant procedures. The staff has determined that these details are not required to be in the TS under 10 CFR 50.36 or Section 182a of the Atomic Energy Act. Further, they do not fall within any of the four criteria discussed in Section 2.0 above. Therefore, the deletion of this TS is acceptable.

3.2.3 Conclusion

The staff finds that the proposed TS Section 3/4.6.B has been reformatted adopting the STS guidelines. The staff has reviewed the proposed TS against the STS and current TS requirements and finds that the relaxation of current TS requirements maintains the design requirements. Therefore, the staff finds proposed TS Section 3/4.6.B acceptable.

3.3 Section 3/4.6.C, Recirculation Pumps

Proposed TS 3/4.6.C, "Recirculation Pumps" incorporates the guidance of the STS Section 3/4.4.1.3 and requirements from Section 3/4.6.H of the current TS for both stations. The proposed TS has been reformatted adopting the STS guidelines. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR.

The proposed action deviates from current requirements by allowing a 2 hour AOT prior to tripping one recirculation pump if the recirculation pump flow limits can not be maintained. These limits require that at >80% power the pump speeds shall be within 10% of each other and at <80% power they shall be within 15% of each other. The purpose of requiring recirculation pump mismatch limits is to comply with the ECCS LOCA analysis. For some limited low probability events with the recirculation loops operating with large speed differences (>15% speed differential at >80% power and >20% speed differential at <80% power), it is possible for the LPCI loop selection logic to select the wrong loop for injection. Therefore, the proposed TS provide a margin of 5% before a problem could arise. Allowing 2 hours to evaluate the problem does not affect the level of safety. In addition, if the pump speeds can be restored within 2 hours, the plant is in a safer condition by avoiding the potential for transients when a pump is tripped and then later restarted.

Because the proposed TS provides more operator guidance and does not change existing safety margins, this change is acceptable.

The staff finds that the proposed TS Section 3/4.6.C, "Recirculation Pumps" has been reformatted adopting the STS guidelines. The staff has reviewed the proposed TS against the STS and current TS requirements and finds that there are no significant deviations from STS and the relaxation of current TS requirements maintains the design requirements. Therefore, the staff finds proposed TS Section 3/4.6.C acceptable.

3.4 Section 3/4.6.D, Idle Recirculation Loop Startup

Proposed TS 3/4.6.D, "Idle Recirculation Loop Startup" incorporates the guidance of the STS Section 3/4.4.1.4 and requirements from Section 3/4.6.H of the current TS for both stations. The proposed TS has been reformatted adopting the STS guidelines. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR.

The staff finds that the proposed TS Section 3/4.6.D, "Idle Recirculation Loop Startup" has been reformatted adopting the STS guidelines. The proposed TS contains an action statement which requires suspension of idle recirculation loop startup if limits are exceeded. Because the current TS do not contain a specific required action, the proposed action is an enhancement of current requirements and is acceptable. The staff has reviewed the proposed TS against the STS and current TS requirements and finds that there are no significant deviations from STS and the proposed TS do not relax any existing TS requirements. Therefore, the staff finds proposed TS Section 3/4.6.D acceptable.

3.5 Section 3/4.6.E, Safety Valves

Proposed TS 3/4.6.E, "Safety Valves" incorporates the guidance of the STS Section 3/4.4.2.1 and requirements from Section 3/4.6.E of the current TS for both stations. Because Dresden and Quad Cities contain separate safety and relief valves as opposed to combination safety/relief valves, the proposed TS is divided into two sections, 3/4.6.E and 3/4.6.F to address the separate valves. The proposed TS has been reformatted adopting the STS guidelines. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR.

The proposed TS adds a requirement that the safety valves be closed with operable position indication. This is an enhancement of current TS which do not provide requirements for position indication. Proposed TS 4.6.E.1 is a new requirement to demonstrate operability of the position indicators. Because the current TS do not contain requirements for position indication, the proposed TS is more conservative and is acceptable.

The proposed TS adds a footnote which clarifies the test conditions for satisfying the TS lift requirements. This footnote is based on STS guidelines

and is an enhancement to current TS by providing additional guidance to operators and is acceptable.

The staff finds that the proposed TS Section 3/4.6.E, "Safety Valves" has been reformatted adopting the STS guidelines. The staff has reviewed the proposed TS against the STS and current TS requirements and finds that there are no significant deviations from STS and the proposed TS do not relax any existing TS requirements. Therefore, the staff finds proposed TS Section 3/4.6.E acceptable.

3.6 Section 3/4.6.F, Relief Valves

Proposed TS 3/4.6.F, "Relief Valves" incorporates the guidance of the STS Section 3/4.4.2.2 and requirements from Sections 3/4.6.E and 3/4.5.D of the current TS for both stations. The proposed TS has been reformatted adopting the STS guidelines. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR.

The current Dresden TS specify a range of acceptable relief valve settings in the LCO. The proposed TS deviates from the current requirement by specifying a maximum setpoint only. The proposed maximum value is equivalent to the current maximum value and is therefore acceptable. The use of a maximum value as opposed to a range of values is consistent with current Quad Cities TS.

The proposed action statements are based on STS guidelines. Action 1 requires that with one or more relief valves open with water temperature less than 110 degrees F, action must be taken to close the valves. If suppression pool temperature is greater than 110 degrees F, the mode switch must be placed in shutdown. The second action is consistent with the requirements of TSUP 3.7.K which specifies 110 degrees as the maximum pool water temperature and requires a shutdown if it is exceeded. Therefore, the proposed action is acceptable.

Current TS 3.5.D.2 allows one relief valve to be out-of-service indefinitely provided that the minimum average planar linear heat generation rate (MAPLHGR) reduction factors are applied to the MAPLHGR limits. The proposed TS do not require that the MAPLHGR reduction factors be applied. However, the proposed TS action 2 is more conservative than the current because it will only allow operation for 14 days before a shutdown is required when one relief valve is out of service rather than indefinite operation. With two relief valves out-of-service, the current TS allow a 7 day AOT. The proposed TS require hot shutdown within 12 hours and cold shutdown within 24 hours. Therefore, the proposed TS is more conservative than the current requirements and is acceptable.

Proposed TS 4.6.F.1.a requires a channel functional test of the relief valve function. In its September 17, 1993, submittal the licensee originally proposed a periodicity of 92 days for functional tests of the relief valves which is more restrictive than the current frequency of every refueling outage. In the June 30, 1995, supplement, the licensee proposed to retain the existing frequency of approximately 18 months based on the guidance of GL

93-05. This will be left as an open item contingent upon review and approval in the clean up amendment.

The proposed TS adds a requirement that the safety valves be closed with operable position indication. This is an enhancement of current TS which do not provide requirements for position indication. Proposed TS 4.6.E.1 is a new requirement to demonstrate operability of the position indicators. Because the current TS do not contain requirements for position indication, the proposed TS is more conservative and is acceptable.

The staff finds that the proposed TS Section 3/4.6.F, "Relief Valves" has been reformatted adopting the STS guidelines. The staff has reviewed the proposed TS against the STS and current TS requirements and finds that the proposed TS do not relax any existing TS requirements. Therefore, the staff finds proposed TS Section 3/4.6.F, with the exception of the open item in TS 4.6.F.1.a, acceptable.

3.7 Section 3/4.6.G, Leakage Detection Systems

Proposed TS 3/4.6.G, "Leakage Detection System" incorporates the guidance of the STS Section 3/4.4.3.1 and requirements from Section 3/4.6.D of the current TS for both stations. The proposed TS has been reformatted adopting the STS guidelines. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR.

The proposed action statements are an enhancement of the current TS by decreasing the AOT from 7 days to 24 hours when the primary containment atmosphere particulate radioactivity sampling system or the drywell floor drain sump system is inoperable. The proposed action is more conservative than the current TS and is acceptable.

The proposed TS SRs are an enhancement of the current TS which only require that the leakage detection systems be observed. The proposed TS require a demonstration of operability. The proposed TS also adds a new requirement for a channel calibration of the drywell floor drain sump pump discharge flow integrator once per 18 months. The proposed TS is more conservative and is, therefore, acceptable.

The staff finds that the proposed TS Section 3/4.6.G, "Leakage Detection System" has been reformatted adopting the STS guidelines. The staff has reviewed the proposed TS against the STS and current TS requirements and finds that there are no significant deviations from STS and the proposed TS enhance the existing TS requirements. Therefore, the staff finds proposed TS Section 3/4.6.G acceptable.

3.8 Section 3/4.6.H, Operational Leakage

Proposed TS 3/4.6.H, "Operational Leakage" incorporates the guidance of the STS Section 3/4.4.3.2 and requirements from Section 3/4.6.D of the current TS for both stations. The proposed TS has been reformatted adopting the STS

guidelines. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR.

3.8.1 LCO

Proposed TS 3.6.H.1 is a new requirement which states that no pressure boundary leakage is acceptable. This new requirement is an enhancement of current TS and is acceptable.

Proposed TS 3.6.H.2 deviates from current TS by specifying that the total allowed leakage be averaged over a 24 hour period. The current TS do not include an allowance for averaging. However, the proposed TS include additional restrictions that limit additional increases in unidentified leakage of greater than or equal to 2 gpm when averaged over a 24 hour period. Unidentified leakage is new leakage above and beyond normal unidentified leakage currently identified as baseline for the plant. This additional restriction ensures new leaks to the reactor coolant system are discovered and appropriate corrective action is initiated. Therefore, the proposed TS provide adequate assurance that increases in leakage are monitored and corrected and provide an equivalent level of safety as the current TS. This change is acceptable.

The proposed requirement which limits the increase in unidentified leakage deviates from STS guidelines by limiting the applicability to Mode 1 only. This limit provides needed flexibility during Mode 2 when leakage rates are increasing to normal baseline levels experienced in Mode 1. Without the revised applicability, reactor operation could not reach Mode 1.

The STS guidance on leakage limits from any reactor coolant system pressure isolation valve, which is based on the guidance of GL 87-06, is not adopted in the proposed TS. In response to the GL, Dresden and Quad Cities outlined the methods currently implemented for assuring the leak tight integrity of all the pressure isolation valves as independent barriers of the reactor coolant system. Neither the Dresden nor Quad Cities design includes high pressure to low pressure interface valve leakage pressure monitors and both plants utilize other existing instrumentation for determination of leakage through a pressure boundary isolation valve. As a result of the design limitations, the STS requirement for reactor coolant system pressure isolation valve leakage limits is not adopted. This deviation is acceptable.

3.8.2 Actions

Proposed ACTION 2 requires that if leakage is greater than the TS limits, the leakage must be restored to within limits in 4 hours or be in hot shutdown in the next 12 hours and cold shutdown in the following 24 hours. This is a relaxation of current TS which do not include an AOT and require cold shutdown in 24 hours. The proposed AOT of 4 hours is consistent with industry practice for restoring leakage rates to within limits and has been shown to provide an adequate level of protection for monitoring leakage to within acceptable levels. The staff has determined that the relaxation of current TS

requirements maintains the design requirements and the proposed action statement is, therefore, acceptable.

Proposed ACTION 3 is a new requirement to determine if the source of unidentified leakage is intergranular stress corrosion cracking. This action is consistent with the guidance in GL 88-01 and is more conservative than current requirements. The LCO limits and associated action statement are applicable regardless of whether unidentified leakage is due to intergranular stress corrosion cracking. Proposed action 3 is an additional requirement which is more restrictive and is acceptable.

3.8.3 Surveillance Requirements

The proposed TS requires sampling of the primary containment atmospheric particulate radioactivity once per 12 hours. This is an enhancement of the current TS which require this sampling once per day.

The proposed TS relax the periodicity of the sump flow monitoring and recording for Dresden Unit 2 from every 4 hours to every 8 hours. The proposed periodicity has been shown, based on industry experience, to provide an adequate level of protection for ensuring plant leakage rates are appropriately monitored and is consistent with the guidance in GL 88-01. An 8 hour surveillance frequency is also consistent with the current TS for Dresden, Unit 3. Therefore, the relaxation in frequency of the sump surveillance has a negligible impact on plant safety margins and is acceptable.

Current Dresden Unit 2 TS 4.6.D.2 has been deleted in the proposed TS. The current TS states that additional leakage limits must be met until the recirculation piping indications have been resolved. The licensee has determined that the recirculation piping indication problems associated with Dresden Unit 2 have been resolved and additional leakage limits requirements are obsolete. The TS requirements associated with GL 88-01 are sufficient for control of leakage from the reactor coolant pressure boundary. Therefore, the deletion of the additional leakage limits is acceptable.

The proposed TS do not include the STS guidelines for leak testing reactor coolant system pressure isolation valves because the associated LCO was not adopted. The proposed TS also do not include the STS guidelines for a channel functional test and calibration of the valve leakage pressure monitors because these requirements are not in current TS and the Dresden and Quad Cities system design is not applicable to the STS requirements. These deviations are acceptable.

3.8.4 Conclusion

The staff finds that the proposed TS Section 3/4.6.H, "Operational Leakage" has been reformatted adopting the STS guidelines. The staff has reviewed the proposed TS against the STS and current TS requirements and finds that all deviations from STS are acceptable and that the relaxation of the current TS

maintains the design requirements for each station. Therefore, the staff finds proposed TS Section 3/4.6.H acceptable.

3.9 Section 3/4.6.I, Chemistry

Proposed TS 3/4.6.I, "Chemistry" incorporates the guidance of the STS Section 3/4.4.4 and requirements from Section 3/4.6.C of the current TS for both stations. The proposed TS has been reformatted adopting the STS guidelines. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR.

3.9.1 LCO

The applicability of proposed TS 3/4.6.I deviates from STS guidelines. STS applicability specifies "At all times" whereas the proposed applicability is Modes 1, 2, and 3. The proposed applicability is based on the current TS which specify chemistry limits during power operation which is Modes 1 and 2. The current TS for Dresden also specify surveillances to be performed until the reactor is in a cold shutdown condition; encompassing Modes 1,2, and 3. Therefore, the proposed applicability is consistent with current TS. To maintain consistency, the proposed TS have also not adopted the STS action statements which are applicable at all times except for Modes 1, 2, and 3.

3.9.2 Actions

Proposed ACTION 1.b is a new requirement adopted from STS guidelines. It requires that if certain levels of conductivity, chloride concentration, or pH are exceeded in Mode 1 for more than 72 hours in one continuous time interval or for more than 336 hours in one year, the plant must be in startup within 8 hours. These requirements are consistent with industry practice and have been shown to provide an adequate level of protection for monitoring residual chemistry effects on the reactor coolant system. Proposed ACTION 1.a allows these levels to be exceeded for less than 72 hours during one continuous interval or less than 336 hours per year without a requirement to report to the Commission. This allows minor deviations in plant chemistry if the excursions are controlled within appropriate levels. In addition, this requirement is an enhancement of current TS in that the current TS do not contain limits for pH. The proposed TS place additional restrictions on the current TS and are acceptable.

Proposed ACTION 1.c is a revision of current Dresden requirements. The current Dresden TS limit for conductivity at steaming rates greater than 100,000 pounds per hour is 5 μmhos/cm. The proposed TS limit is 10 μmhos/cm in Mode 1. The current limit for Quad Cities is 10 μmho/cm and, therefore, there is no change. The proposed TS contain the additional requirement in action 1.b that if conductivity exceeds 1.0 μmhos/cm for more than 72 hours, the plant is required to be in startup in 8 hours. The additional limit of 1.0 μmho/cm is much more restrictive than the current Dresden TS limit of 5 μmhos/cm and will provide adequate indication when minor deviations in plant chemistry occur. This additional requirement will assurance that minor

deviations in plant chemistry are controlled and provides a sound technical basis for the revision of the Dresden TS limit. Therefore, this change is acceptable.

The proposed TS deviate from the current requirements for conductivity in Modes 2 and 3. The current TS state that the conductivity limit during the first 24 hours of a reactor startup is 10 μmho/cm. The proposed TS applicability is based on operational modes. The proposed TS limit during modes 2 and 3 (startup) is 2.0 μmho/cm with the allowance that no action needs to be taken until this limit is exceeded for greater than 48 hours. Although the 48 hour allowance is less conservative than the current TS time allowance, the more conservative TS limit (2.0 μmho/cm versus the current 10 μmho/cm) compensates for this and ensures that the plant is operating in a safer condition. Therefore, the proposed TS is acceptable.

The current limit for chloride concentration during the first 24 hours following a reactor startup is 0.1 ppm. This limit is more restrictive than the limit during power operation to avoid exceeding the adverse chloride-oxygen combinations which result from the higher content of dissolved oxygen in the reactor coolant during startups. The proposed TS also limit the chloride concentration to 0.1 ppm in Modes 2 and 3. In accordance with STS guidance, the proposed TS allow this limit to be exceeded for up to 48 hours. The proposed 48 hour allowance is a reasonable period of time to monitor chloride levels and determine the cause of the deviation prior to taking action. Therefore, this change is acceptable.

3.9.3 Surveillance Requirements

The proposed TS enhance the current SR for Mode 1. The current TS require measurement every 96 hours while the proposed TS require measurement of chemistry and chloride levels every 72 hours. The proposed TS are more conservative and are, therefore, acceptable.

The proposed TS decreases the frequency of analyzing reactor coolant in Modes 2 and 3. The current TS require analysis of conductivity and chloride concentration every 4 hours in these modes. The proposed TS require an analysis every 72 hours. However, the proposed TS also require continuous recording of the conductivity of the reactor coolant. In the event the continuous monitor is inoperable, the proposed TS require in-line measurements every 4 hours. Also, if conductivity levels are greater than those specified in proposed Table 3.6.I-1 (which are more restrictive than the limits in the current TS), an analysis must be made every 8 hours. The proposed frequencies provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action. The proposed frequency has been demonstrated based on industry experience to adequately monitor plant chemistry limits and does not significantly reduce existing plant safety margins. Therefore, the proposed TS is acceptable.

The proposed TS increase the required frequency of conductivity measurements when the continuous monitor is inoperable. The current TS require measurement

once per day, while the proposed TS require a conductivity measurement once per 4 hours. This is an enhancement of the current TS and is acceptable.

The proposed TS eliminate the current requirement for chloride ion content measurements once per day whenever the continuous conductivity monitor is inoperable. This is a relaxation of current requirements but does not significantly impact plant safety because the proposed TS contain other more restrictive requirements which compensate for the relaxation this requirement. As discussed in the paragraph above, if the monitor is inoperable then conductivity must be measured once per 4 hours. If the measurement for conductivity determines that the level exceeds the allowable, then proposed TS 4.6.I.2.2 requires that a sample be analyzed for chloride every 8 hours. The proposed TS provides adequate assurance that the conductivity and chloride levels are within limits and is acceptable.

The proposed TS include a new requirement to measure pH every 8 hours whenever conductivity is greater than the TS limits. This requirement is not in the current TS, but is consistent with the proposed LCO which requires pH to be within specified limits. This requirement deviates from the STS guideline to analyze pH every 72 hours. The STS requirement was not adopted in the proposed TS because when the conductivity is within limits, the pH, chlorides and other impurities affecting conductivity must also be within their acceptable limits. The proposed TS is more conservative than the current TS and is, therefore, acceptable.

The proposed TS include a new requirement to measure chemistry limits prior to pressurizing the reactor during each startup. This is an enhancement of the current TS and is acceptable.

3.9.4 Conclusion

The staff finds that the proposed TS Section 3/4.6.I, "Chemistry" has been reformatted adopting the STS guidelines. The staff has reviewed the proposed TS against the STS and current TS requirements and finds that all deviations from STS are acceptable and that any relaxations of the current TS non-radioactive chemistry requirements are sufficiently compensated by more restrictive proposed TS requirements or will not significantly change the design requirements for each station. Therefore, the staff finds proposed TS Section 3/4.6.I acceptable.

3.10 <u>Section 3/4.6.J, Specific Activity</u>

Proposed TS 3/4.6.J, "Specific Activity" incorporates the guidance of the STS Section 3/4.4.5 and requirements from Section 3/4.6.C of the current TS for both stations. The proposed TS has been reformatted adopting the STS guidelines. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR.

3.10.1 LCO

The current Quad Cities TS specify a limit for radioiodine concentration of 5 μCi/gm of I-131. The proposed TS contain a more restrictive limit of 0.2 μCi/gm. Therefore, the proposed TS are more conservative and are acceptable. The current Dresden TS specify a limit of 0.2 μCi/gm and is unchanged in the proposed TS.

The proposed TS did not adopt the STS guidance to maintain the average disintegration energy below a certain limit. The proposed TS also eliminate the associated SR. On-line monitoring capability at Dresden and Quad Cities Station eliminates the need for this requirement. In addition, the current TS do not contain these requirements. The proposed TS does not relax any current TS requirements and is, therefore, acceptable.

3.10.2 Actions

Proposed action 1 deviates from current TS when the specific activity exceeds the TS limits. The current Dresden TS require immediate initiation of a shutdown and the reactor to be in cold shutdown within 24 hours. The current Quad Cities TS require that an orderly shutdown be initiated. The proposed TS require that the plant be brought to hot shutdown with the main steam isolation valves (MSIVs) closed within 12 hours. The proposed TS place the plant in the appropriate mode for which reactor coolant activity concerns are negligible in a shorter time period than the current TS. Closing the MSIVs is a new requirement which prevents the release of activity to the environs should a steamline rupture occur outside containment. The requirement to go to hot shutdown as opposed to cold shutdown provides an adequate level of protection and is acceptable.

The proposed TS enhance the current Dresden TS by eliminating the allowance to perform a second sample within 8 hours when the activity is greater than 4.0 μCi/gm (the current Quad Cities TS do not contain this allowance). The proposed TS require immediate action to be taken when a measurement of greater than 4.0 uCi/gm is taken rather than waiting 8 hours to perform a second sample. The proposed TS are more conservative and are, therefore, acceptable.

The current Dresden TS contain a requirement to analyze reactor coolant samples every 8 hours during plant shutdown when activity is greater than 4.0 μCi/gm. This requirement has been deleted in the proposed TS consistent with the proposed action. The proposed action in this circumstance is to take the plant to hot shutdown in 12 hours and not to cold shutdown. When the activity is greater than 4.0 μCi/gm the proposed TS require sampling in accordance with proposed Table 4.6.J-1, Item 3.a: sampling every 4 hours. Therefore, the proposed TS are more restrictive and the additional requirement to sample every 8 hours is unnecessary. The proposed action provides an adequate level of protection without the additional sampling requirements and is acceptable.

The proposed TS enhance the current requirements for additional sampling if the TS limit is exceeded. The current Dresden TS require that, when specific activity is greater than 0.2 μCi/gm, an analysis must be performed three times every 24 hours. The current Quad Cities TS are discussed below. The proposed TS require a surveillance every 4 hours until the limit is restored. The proposed TS are more conservative than the current TS and are acceptable.

Current Quad Cities TS 4.6.C.1.c requires sampling 24 hours prior to reactor startups when steady-state iodine concentrations are greater than 1 percent, but less than 10 percent of the TS limits (0.05 μCi/gm, but less than 0.5 μCi/gm). The proposed TS requires sampling and analysis when specific activity is greater than 0.2 μCi/gm dose equivalent I-131. The current TS is unclear as it places a limit on specific activity prior to reactor startup that is based on previous levels of activity. The current requirement to ensure the affected limits are within acceptance levels prior to reactor startups is based on the fact that radioiodine concentration can change rapidly in the reactor coolant during transient reactor operations, such as startups, if failed fuel is present. Although reactor coolant sampling is ineffective as a means to rapidly detect gross fuel element failures, some capability to detect gross fuel element failures is inherent in the radiation monitors in the off-gas system. The current TS provides a method to detect changes in radioiodine concentration which may have occurred during previous periods of power and/or shutdown operations. These sampling requirements provide escalating sampling criteria during a reactor startup based upon previous operational radioiodine concentration. The need for these sampling criteria are based on early BWR fuel failure experience. Since that time, fuel performance at Quad Cities, and throughout the industry, has improved to the point that a pre-operational check of reactor coolant radioiodine concentration is no longer necessary. As such, the proposed and STS sampling requirements are based on changes in power level and/or offgas radiation levels. The proposed TS provides clearer guidance to operators. The requirement to ensure that levels are acceptable prior to performing a mode change is encompassed within 4.0.D which does not allow a mode change unless the SRs for that mode have been performed. Based on the above discussion, the proposed TS is acceptable.

Proposed ACTION 3 requires that additional sampling should be performed when there is a rapid power change or offgas change. These are new requirements for Dresden and are similar to current Quad Cities TS. The proposed TS is more conservative than the current Dresden TS and is an enhancement of the current Quad Cities TS because it provides more explicit guidance to operators.

3.10.3 Surveillance Requirements

The current Dresden TS require an analysis of total activity (beta and gamma) content every 96 hours. The current Quad Cities TS do not contain a requirement to measure activity. The proposed TS increase the frequency of beta and gamma activity measurement to every 72 hours. This is an enhancement of the current requirements for Dresden and adds new requirements for Quad

Cities. The proposed TS is more conservative than current TS and is acceptable.

The current TS require an analysis of dose equivalent I-131 every 96 hours. The proposed TS decrease the frequency of dose equivalent I-131 measurement from 96 hours to every 31 days. This decrease in frequency is acceptable based on the fact that the frequency of gross beta/gamma activity measurement is increased. Gross increases in beta/gamma activity should act as precursor to any potential dose equivalent I-131 excursions. Therefore, the decrease in frequency of dose equivalent I-131 measurement has an insignificant impact on safety and is acceptable.

The proposed TS contain a requirement to perform an isotopic analysis of an off-gas sample every 31 days. This is a new requirement for Dresden and incorporates the requirements of the current Quad Cities TS. The proposed TS is an enhancement of current requirements and is acceptable.

3.10.4 Conclusion

The staff finds that the proposed TS Section 3/4.6.J, "Specific Activity" has been reformatted adopting the STS guidelines. The staff has reviewed the proposed TS against the STS and current TS requirements and finds that there are no significant deviations from STS and that the relaxation of the current TS maintains the design requirements for each station. Therefore, the staff finds proposed TS Section 3/4.6.J acceptable.

3.11 Section 3/4.6.K, Pressure/Temperature Limits

Proposed TS 3/4.6.K, "Pressure/Temperature Limits" incorporates the requirements of the STS Section 3/4.4.6.1 and requirements from Section 3/4.6.A and 3/4.6.B of the current TS for both stations. The proposed TS has been reformatted adopting the STS guidelines. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR.

3.11.1 LCO

The current TS requirement which allows a step reduction of 240 degrees Fahrenheit has been deleted in the proposed TS. This uncontrolled cool down rate is based on the maximum expected transient over the lifetime of the reactor and was considered in the design of the pressure vessel. This requirement is a design detail which is not appropriate for control within the TS. This detail is controlled in an administrative program for tracking vessel thermal transients. The staff has determined that the requirements for a step reduction in reactor coolant system temperature are not required to be in the TS under 10 CFR 50.36 or Section 182a of the Atomic Energy Act. Further, they do not fall within any of the four criteria discussed in Section 2.0, above. Therefore, the deletion of this TS is acceptable.

The current TS requirement that the shell flange to shell temperature differential must be less than 140 degrees Fahrenheit has been deleted in the

proposed TS. The analyses considered in the design of the pressure vessel used a heating and cooling rate of 100 degrees Fahrenheit/hour which is controlled by proposed TS 3.6.K.1 and 3.6.K.2. The flange metal temperature differential of 140 degrees Fahrenheit was determined as a result of these analyses. Therefore, a heatup/cooldown rate of 100 degrees Fahrenheit/hour should ensure that the temperature differential of 140 degrees Fahrenheit will not be challenged. Maintaining the heatup/cooldown rates will provide an adequate level of safety without the need for additional restrictions on the differential temperature. Therefore, this TS is a design requirement more appropriately controlled within the plants' UFSARs. The staff has determined that the requirements for shell flange to shell temperature differential are not required to be in the TS under 10 CFR 50.36 or Section 182a of the Atomic Energy Act. Further, they do not fall within any of the four criteria discussed in Section 2.0, above. In addition, the staff finds that sufficient regulatory controls exist under 10 CFR 50.59 to assure continued protection of public health and safety.

Proposed TS 3.6.K.3 is a new requirement which includes specific limitations on the maximum reactor coolant temperature change during hydrostatic and leak testing operations. The proposed TS is an enhancement of current requirements and is acceptable.

Current TS 3.6.B.2 for Dresden, Unit 2, requires that the temperature of the vessel shell be greater than or equal to 80 degrees Fahrenheit for the reactor vessel head bolting studs to be under tension. The current TS contain a discrepancy between the value in TS 3.6.B.2 (80 degrees Fahrenheit) and the value in TS Figure 3.6.1 (100 degrees Fahrenheit). The 80 degrees Fahrenheit minimum boltup temperature within the Unit 2 current TS was consistent with an earlier version of current TS Figure 3.6.1. However, the licensee's review of GL 92-01 required a revision to Figure 3.6.1 and the associated LCO. The revised curves reflecting 100 degrees Fahrenheit were approved in Amendment No. 123; however, the corresponding revision to TS 3.6.B.2 was inadvertently omitted. The licensee has identified this issue and controls the requirement under administrative measure. Proposed TS 3.6.K.4 uses the correct value of 100 degrees Fahrenheit. The current Quad Cities TS uses the value of 100 degrees, and therefore, there is no change in the proposed TS.

3.11.2 Actions

The proposed actions are adopted from STS guidelines since the current TS do not specify explicit actions. The proposed action requirements include time limitations for evaluating potentially degraded conditions and for performing appropriate actions. These actions are an enhancement of current TS and are acceptable.

3.11.3 Surveillance Requirements

The proposed TS relax the current requirement to permanently record temperatures and pressure at 15 minute intervals. The proposed TS requires temperatures to be measured every 30 minutes. The proposed TS provides an

adequate frequency to monitor heatups and cooldowns. The proposed reduction in the periodicity has a negligible effect on existing plant safety margins and is acceptable. The requirement to permanently record the temperatures and pressures has not been retained in proposed TS section 3.6.K, but is encompassed within proposed TS Section 6.0 which will require that the licensee retain the records of all TS surveillance actions for five years and the records of transient and operational cycles for the duration of the unit operating license. Proposed TS 6.0 will be reviewed separately. This issue will be an open item pending staff approval of TSUP Section 6.0.

The current TS include details regarding the location at which temperatures must be recorded and the requirement that these be recorded until three consecutive readings are within 5 degrees of each other. The details for the methods of performing SRs are inappropriate for inclusion within the TS and have been deleted. The details are controlled by procedures. This change does not change the surveillances to be performed and does not relax any current TS requirements. The staff has determined that these details are not required to be in the TS under 10 CFR 50.36 or Section 182a of the Atomic Energy Act. Further, they do not fall within any of the four criteria discussed in Section 2.0 above. Therefore, the deletion of this TS is acceptable.

The current TS contain details regarding the installation, removal, and testing of neutron flux monitors and samples. These specific details related to the methods for performing surveillances are inappropriate for inclusion in the TS and are adequately controlled by procedures. The proposed TS also delete the current TS table which provides the schedule for removing samples. This is a deviation from the STS guidelines, but is consistent with the guidance given in GL 91-01, "Removal of the Schedule for the Removal of Reactor Vessel Material Specimens From the Technical Specifications." The schedule is contained in plant procedures. The current TS state that the testing of samples will be used to determine nil ductility transition temperature (NDTT). This requirement is encompassed by the 10 CFR Part 50, Appendix H, requirements to periodically generate the curves of TSUP Figure 3.6.K-1. Therefore, retention of a separate requirement to determine the NDTT would be redundant and has been deleted. The staff has determined that these requirements are not required to be in the TS under 10 CFR 50.36 or Section 182a of the Atomic Energy Act. Further, they do not fall within any of the four criteria discussed in Section 2.0 above. These changes are acceptable.

3.11.4 Conclusion

The staff finds that the proposed TS Section 3/4.6.K, "Pressure/Temperature Limits" has been reformatted adopting the STS guidelines. The staff has reviewed the proposed TS against the STS and current TS requirements and finds that there are no significant deviations from STS and that the relaxation of the current TS maintains the design requirements for each station. Therefore, the staff finds proposed TS Section 3/4.6.K acceptable.

3.12 Section 3/4.6.L, Reactor Steam Dome

Proposed TS 3/4.6.L, "Reactor Steam Dome" is a new TS which incorporates the guidance of STS Section 3/4.6. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR. The proposed requirements are applicable to the Dresden and Quad Cities plant design and are an enhancement of the current TS. The proposed TS is, therefore, acceptable.

3.13 Section 3/4.6.M, Main Steamline Isolation Valves

Proposed TS 3/4.6.M, "Main Steamline Isolation Valves" is a new TS which incorporates the guidance of the STS Section 3/4.4.7. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR. The proposed requirements are applicable to the Dresden and Quad Cities plant design and are an enhancement of the current TS. The proposed TS is, therefore, acceptable.

3.14 Section 3/4.6.N, Structural Integrity

Proposed TS 3/4.6.N, "Structural Integrity" incorporates the guidance of the STS Section 3/4.4.8 and requirements from Section 3/4.6.F of the current TS for both stations. The proposed TS has been reformatted adopting the STS guidelines. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR.

The current TS contain detailed information from Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) which has been deleted in the proposed TS. The proposed TS requires that structural integrity be maintained in accordance with TSUP Section 4.0.E which defines the applicability of the ASME Code, Section XI. Therefore, the details from the ASME Code are not necessary for inclusion in the TS and may be deleted.

The current TS do not include specific actions to take if the primary system boundary has indications in excess of the allowables. Therefore, the proposed actions are new requirements, adopted from STS guidelines, which are an enhancement of the current TS and are acceptable.

The staff finds that the proposed TS Section 3/4.6.N, "Structural Integrity" has been reformatted adopting the STS guidelines. The staff has reviewed the proposed TS against the STS and current TS requirements and finds that there are no significant deviations from STS and no relaxations of the current TS. Therefore, the staff finds proposed TS Section 3/4.6.N acceptable.

3.15 Section 3/4.6.O, Shutdown Cooling - Hot Shutdown (Dresden)
Section 3/4.6.O, Residual Heat Removal - Hot Shutdown (Quad Cities)
Section 3/4.6.P, Shutdown Cooling- Cold Shutdown (Dresden)
Section 3/4.6.P, Residual Heat Removal - Cold Shutdown (Quad Cities)

Proposed TS 3/4.6.O and 3/4.6.P incorporate the guidance of the STS Sections 3/4.4.9.1 and 3/4.4.9.2. These are new TS sections not contained in current TS. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR. The proposed TS for Dresden and Quad Cities are slightly different due to the different systems used for shutdown cooling purposes. Dresden has a separate shutdown cooling system and Quad Cities uses the residual heat removal (RHR) system.

The proposed TS for Quad Cities deviate from the STS guidelines and from the proposed Dresden TS due to the design of the RHR system at Quad Cities. Quad Cities' RHR system configuration does not allow either the shutdown cooling flow or the service water cooling flow to be throttled sufficiently to maintain constant temperature. Therefore, the proposed Quad Cities TS require that one subsystem be operable, as opposed to the proposed Dresden TS requirement and STS guidance to be in operation. This deviation is consistent with plant design and is acceptable.

The proposed TS are applicable to the Dresden and Quad Cities plant designs and provide enhanced guidance to operations personnel. Therefore, the proposed TS are acceptable.

3.16 Current TS Section 3/4.6.I, Snubbers

Current TS 3/4.6.I, "Snubbers" has been relocated to proposed TS 3/4.8.F. The changes to this section are based on STS guidelines and the guidelines of GL 90-09, "Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions." Proposed TS 3/4.8.F will be reviewed separately and this issue will remain an open item pending NRC staff approval of proposed TS 3/4.8.

3.17 Bases

The staff has reviewed the proposed Bases for TSUP TS Section 3/4.6. The proposed Bases have been prepared using the guidelines of the STS. The staff finds the proposed Bases acceptable with the following comment. The last paragraph on page B 3/4.6-7 discusses evaluating reactor vessel material specimens. The correct standard which should be referenced is ASTM E 185-73. This will be an open item pending its correction in the clean-up amendment.

3.18 Open Items

The following items should be left as open items contingent upon their disposition in the cleanup amendment.

1. The number of jet pumps in 3.6.B Action 2 will be revised.

2. The periodicity of TS 4.6.F.1.a (relief valve testing) will be revised.

3. Bases page B 3/4.6-7 will be corrected to reference the ASTM standard.

4. Current TS 3/4.6.I will be relocated to proposed TS 3/4.8.F.

5. Current TS 4.6.A and 4.6.B regarding permanent recording of reactor vessel temperatures and reactor coolant pressures will be located in proposed TS section 6.0.

4.0 SUMMARY

The proposed TS for Section 3/4.6 will be clearer and easier to use as a result of the adoption of the STS format. The changes result in additional limitations, restrictions, or changes based on generic guidance. It is the staff's assessment that the changes proposed in this amendment do not pose any decrease in safety, or an increase in the probability of an analyzed or unanalyzed accident. The revised TS changes do not reduce the existing margin of safety set forth by the current TS. Therefore, the staff finds the proposed TS changes acceptable.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendments. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (60 FR 37087). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: D. Skay

Date: September 21, 1995

The application dated September 17, 1993, as supplemented June 30, 1995, contains the proposed upgrade of Section 3/4.6 (Primary System Boundary) of the Dresden and Quad Cities TS.

The review guidance to be used by the NRC staff in the review of the TSUP is described in Section 2.0 of the enclosed Safety Evaluation (SE). The staff reviewed the proposed changes and evaluated all deviations and changes between the proposed TS, the STS, and the current TS.

Based on discussions between ComEd and the staff, it has been mutually agreed upon that the NRC will review the sections of TSUP as they are submitted and provide ComEd an amendment for each submittal. Once all of the TSUP sections have been reviewed and the amendments issued, it is our understanding that ComEd will make one final submittal addressing any changes that may be required as a result of problems uncovered during the course of this effort. Upon receipt and review of this final submittal, the staff will issue a final amendment which addresses any remaining open items and any changes or corrections to the previous amendments. The applicable TSUP TS will be issued with each amendment and will become effective no later than December 31, 1995, for Dresden and June 30, 1996, for Quad Cities.

The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by:

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

Docket Nos. 50-237, 50-249, 50-254, 50-265

Enclosures:
1. Amendment No. 140 to DPR-19
2. Amendment No. 134 to DPR-25
3. Amendment No. 162 to DPR-29
4. Amendment No. 158 to DPR-30
5. Safety Evaluation

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