DOCKET FILE

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

July 27, 1995

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

SUBJECT: ISSUANCE OF AMENDMENTS RELATED TO TSUP SECTION 3.3 (TAC NOS. M92928, M92929, M84319, and M84320)

Dear Mr. Farrar:

The Commission has issued the enclosed Amendment No. 137 to Facility Operating License No. DPR-19 and Amendment No. 131 to Facility Operating License No. DPR-25 for the Dresden Nuclear Power Station, Units 2 and 3, respectively; and Amendment No. 158 to Facility Operating License No. DPR-29 and Amendment No. 154 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 July 29, 1992, as supplemented by letters dated January 14, 1993, February 16, 1993, and May 9, 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requirements such as limiting conditions for operation and action statements utilizing STS terminology, (3) deleting superseded requirements and

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modifications to the TS based on the licensee's responses to Generic Letters, and (4) relocating specific items to more appropriate TS locations.

The application dated July 29, 1992, as supplemented January 14, 1993, February 16, 1993 and May 9 1995, contains the proposed upgrade of Section 3.3 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. 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. 137 to DPR-19
2. Amendment No. 131 to DPR-25
3. Amendment No. 158 to DPR-29
4. Amendment No. 154 to DPR-30
5. Safety Evaluation

cc w/encls: see next page

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DOCUMENT NAME:DRESDEN\DRQC84319.AMD * See previous concurrence

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TS pages not reviewed.

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D. L. Farrar
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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. 137
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 July 29, 1992, as supplemented by letters dated January 14, 1993, February 16, 1993, and May 9, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 137 , 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

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: July 27, 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. 131
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 July 29, 1992, as supplemented by letters dated January 14, 1993, February 16, 1993, and May 9, 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. 131 , 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

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: July 27, 1995

ATTACHMENT TO LICENSE AMENDMENT NOS. 137 AND 131

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
3/4.3-1	3/4.3-1	3/4.3-1
3/4.3-2	3/4.3-2	3/4.3-2
3/4.3-3	3/4.3-3	3/4.3-3
3/4.3-4	3/4.3-4	3/4.3-4
3/4.3-5	3/4.3-5	3/4.3-5
3/4.3-6	3/4.3-6	3/4.3-6
3/4.3-7	3/4.3-7	3/4.3-7
3/4.3-8	3/4.3-8	3/4.3-8
3/4.3-9	3/4.3-9	3/4.3-9
3/4.3-10	3/4.3-10	3/4.3-10
3/4.3-11	3/4.3-11	3/4.3-11
3/4.3-12	3/4.3-12	3/4.3-12
3/4.3-13	3/4.3-13	3/4.3-13
---	---	3/4.3-14
---	---	3/4.3-15
---	---	3/4.3-16
---	---	3/4.3-17
---	---	3/4.3-18
---	---	3/4.3-19
---	---	3/4.3-20
B 3/4.3-14	B 3/4.3-14	B 3/4.3-1
B 3/4.3-15	B 3/4.3-15	B 3/4.3-2
B 3/4.3-16	B 3/4.3-16	B 3/4.3-3
B 3/4.3-17	B 3/4.3-17	B 3/4.3-4
B 3/4.3-18	B 3/4.3-18	B 3/4.3-5
B 3/4.3-19	B 3/4.3-19	B 3/4.3-6
B 3/4.3-20	B 3/4.3-20	B 3/4.3-7
B 3/4.3-21	B 3/4.3-21	---
B 3/4.3-22	B 3/4.3-22	---

3.3 - LIMITING CONDITIONS FOR OPERATION

A. SHUTDOWN MARGIN (SDM)

The SHUTDOWN MARGIN (SDM) shall be equal to or greater than:

1. 0.35% $\Delta k/k$ with the highest worth control rod analytically determined, or
2. 0.25% $\Delta k/k$ with the highest worth control rod determined by test.

APPLICABILITY:

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

ACTION:

With the SHUTDOWN MARGIN less than specified:

1. In OPERATIONAL MODE 1 or 2, restore the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.
2. In OPERATIONAL MODE 3 or 4, immediately verify all insertable control rods to be fully inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL MODE 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.
3. In OPERATIONAL MODE 5, suspend CORE ALTERATION(s) and other activities that could reduce the SHUTDOWN MARGIN and fully insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

4.3 - SURVEILLANCE REQUIREMENTS

A. SHUTDOWN MARGIN

The SHUTDOWN MARGIN shall be determined to be equal to or greater than that specified at any time during the operating cycle:

1. By demonstration, prior to or during the first startup after each refueling outage.
2. Within 24 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or known to be unscrammable. The required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or unscrammable control rod.
3. By calculation, prior to each fuel movement during the fuel loading sequence.

3.3 - LIMITING CONDITIONS FOR OPERATION

B. Reactivity Anomalies

The reactivity equivalence of the difference between the actual critical control rod configuration and the predicted control rod configuration shall not exceed 1% Δk/k.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

With the reactivity equivalence difference exceeding 1% Δk/k, within 12 hours perform an analysis to determine and explain the cause of the reactivity difference; operation may continue if the difference is explained and corrected.

With the provisions of the ACTION above not met, be in at least HOT SHUTDOWN within the next 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

B. Reactivity Anomalies

The reactivity equivalence of the difference between the actual critical control rod configuration and the predicted control rod configuration shall be verified to be less than or equal to 1% Δk/k:

1. During the first startup following CORE ALTERATION(s), and
2. At least once per 31 effective full power days.

3.3 - LIMITING CONDITIONS FOR OPERATION

C. Control Rod OPERABILITY

All control rods shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

1. With one control rod inoperable due to being immovable as a result of excessive friction or mechanical interference, or known to be unscrammable:
   a. Within one hour:
      1) Verify that the inoperable control rod, if withdrawn, is separated from all other inoperable withdrawn control rods by at least two control cells in all directions.
      2) Disarm the associated directional control valves[a] either:
         a) Electrically, or
         b) Hydraulically by closing the drive water and exhaust water isolation valves.
   b. With the provisions of ACTION 1.a above not met, be in at least HOT SHUTDOWN within the next 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

C. Control Rod OPERABILITY

1. When above the low power setpoint of the RWM, all withdrawn control rods not required to have their directional control valves disarmed electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch:
   a. At least once per 7 days, and
   b. At least once per 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference, or known to be unscrammable.
2. All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.3.D, 4.3.F, 4.3.G, 4.3.H and 4.3.I.

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a May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

c. Comply with Surveillance Requirement 4.3.A.2 within 24 hours or be in HOT SHUTDOWN within the next 12 hours.

2. With one or more control rods scrammable but inoperable for causes other than addressed in ACTION 3.3.C.1 above:

   a. If the inoperable control rod(s) is withdrawn, within one hour:

      1) Verify that the inoperable withdrawn control rod(s) is separated from all other inoperable withdrawn control rods by at least two control cells in all directions, and

      2) Demonstrate the insertion capability of the inoperable withdrawn control rod(s) by inserting the inoperable withdrawn control rod(s) at least one notch by drive water pressure within the normal operating range.[b]

   b. With the provisions of ACTION 2.a above not met, fully insert the inoperable withdrawn control rod(s) and disarm the associated directional control valves[a] either:

      1) Electrically, or

      2) Hydraulically by closing the drive water and exhaust water isolation valves.

4.3 - SURVEILLANCE REQUIREMENTS

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b The inoperable control rod may then be withdrawn to a position no further withdrawn than its position when found to be inoperable.

a May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

c. If the inoperable control rod(s) is fully inserted, within one hour disarm the associated directional control valves[a] either:

   1) Electrically, or

   2) Hydraulically by closing the drive water and exhaust water isolation valves.

3. With the provisions of ACTION 2 above not met, be in at least HOT SHUTDOWN within the next 12 hours.

4. With more than 8 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

a May be reamed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

D. Maximum Scram Insertion Times

The maximum scram insertion time of each control rod from the fully withdrawn position to 90% insertion, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7 seconds.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding 7 seconds:

1. Declare the control rod(s) exceeding the above maximum scram insertion time inoperable, and
2. When operation is continued with three or more control rods with maximum scram insertion times in excess of 7 seconds, perform Surveillance Requirement 4.3.D.3 at least once per 60 days of POWER OPERATION.

With the provisions of the ACTION(s) above not met, be in at least HOT SHUTDOWN within 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

D. Maximum Scram Insertion Times

The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than 800 psig and, during single control rod scram time tests, with the control rod drive pumps isolated from the accumulators:

1. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER:
   a. following CORE ALTERATION(s), or
   b. after a reactor shutdown that is greater than 120 days,
2. For specifically affected individual control rods[a] following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and
3. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of POWER OPERATION.

a The provisions of Specification 4.0.D are not applicable provided this surveillance is conducted prior to exceeding 40% of RATED THERMAL POWER.

3.3 - LIMITING CONDITIONS FOR OPERATION

E. Average Scram Insertion Times

The average scram insertion time of all OPERABLE control rods from the fully withdrawn position, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.900
50	2.00
90	3.50

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

With the average scram insertion time exceeding any of the above limits, be in at least HOT SHUTDOWN within 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

E. Average Scram Insertion Times

The control rod average scram times shall be demonstrated by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.3.D.

3.3 - LIMITING CONDITIONS FOR OPERATION

F. Group Scram Insertion Times

The average of the scram insertion times, from the fully withdrawn position, for the three fastest control rods of all groups of four control rods in a two-by-two array, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.120
90	3.800

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

With the average scram insertion times of control rods exceeding the above limits:

1. Declare the control rods exceeding the above average scram insertion times inoperable until an analysis is performed to determine that required scram reactivity remains for the slow four control rod group, and
2. When operation is continued with an average scram insertion time(s) in excess of the average scram insertion time limit, perform Surveillance Requirement 4.3.D.3 at least once per 60 days of POWER OPERATION.

With the provisions of the ACTION(s) above not met, be in at least HOT SHUTDOWN within 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

F. Group Scram Insertion Times

All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.3.D.

3.3 - LIMITING CONDITIONS FOR OPERATION

G. Control Rod Scram Accumulators

All control rod scram accumulators shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 5[a].

ACTION:

1. In OPERATIONAL MODE 1 or 2:
    a. With one control rod scram accumulator inoperable, within 8 hours:
        1) Restore the inoperable accumulator to OPERABLE status, or
        2) Declare the control rod associated with the inoperable accumulator inoperable.
    b. With the provisions of ACTION 1.a above not met, be in at least HOT SHUTDOWN within the next 12 hours.
    c. With more than one control rod scram accumulator inoperable, declare the associated control rods inoperable and:

4.3 - SURVEILLANCE REQUIREMENTS

G. Control Rod Scram Accumulators

Each control rod scram accumulator shall be determined OPERABLE at least once per 7 days by verifying that the indicated pressure is ≥800 psig unless the control rod is fully inserted and disarmed, or scrammed.

a In OPERATIONAL MODE 5, this Specification is applicable for the accumulators associated with each withdrawn control rod and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

3.3 - LIMITING CONDITIONS FOR OPERATION

1) If the control rod associated with any inoperable scram accumulator is withdrawn, immediately verify that at least one control rod drive pump is operating by inserting at least one withdrawn control rod at least one notch. With no control rod drive pump operating, immediately place the reactor mode switch in the Shutdown position.

2) Fully insert the inoperable control rods and disarm the associated directional control valves[b] either:

a) Electrically, or

b) Hydraulically by closing the drive water and exhaust water isolation valves.

d. With the provisions of ACTION 1.c.2 above not met, be in at least HOT SHUTDOWN within 12 hours.

2. In OPERATIONAL MODE 5[a]:

a. With one withdrawn control rod with its associated scram accumulator inoperable, fully insert the affected control rod and disarm the associated directional control valves[b] within one hour, either:

4.3 - SURVEILLANCE REQUIREMENTS

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a In OPERATIONAL MODE 5, this Specification is applicable for the accumulators associated with each withdrawn control rod and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

b May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

4.3 - SURVEILLANCE REQUIREMENTS

1) Electrically, or

2) Hydraulically by closing the drive water and exhaust water isolation valves.

b. With more than one withdrawn control rod with the associated scram accumulator inoperable or no control rod drive pump operating, immediately place the reactor mode switch in the Shutdown position.

3.3 - LIMITING CONDITIONS FOR OPERATION

H. Control Rod Drive Coupling

All control rods shall be coupled to their drive mechanisms.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 5[a].

ACTION:

1. In OPERATIONAL MODE 1 or 2 with one control rod not coupled to its associated drive mechanism, within 2 hours:

   a. If permitted by the RWM, insert the control rod drive mechanism to accomplish recoupling and verify recoupling by withdrawing the control rod, and:

      1) Observing any indicated response of the nuclear instrumentation, and

      2) Demonstrating that the control rod will not go to the overtravel position.

   b. If not permitted by the RWM or, if recoupling is not accomplished in accordance with ACTION 1.a above, then declare the control rod inoperable, fully insert the control rod and disarm the associated directional control valves[b] either:

      1) Electrically, or

4.3 - SURVEILLANCE REQUIREMENTS

H. Control Rod Drive Coupling

Each affected control rod shall be demonstrated to be coupled to its drive mechanism by observing any indicated response of the nuclear instrumentation while withdrawing the control rod to the fully withdrawn position and then verifying that the control rod drive does not go to the overtravel position:

1. Prior to reactor criticality after completing CORE ALTERATION(s) that could have affected the control rod drive coupling integrity,

2. Anytime the control rod is withdrawn to the "Full out" position in subsequent operation, and

3. Following maintenance on or modification to the control rod or control rod drive system which could have affected the control rod drive coupling integrity.

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a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

b May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

4.3 - SURVEILLANCE REQUIREMENTS

2) Hydraulically by closing the drive water and exhaust water isolation valves.

2. With the provisions of ACTION 1 above not met, be in at least HOT SHUTDOWN within 12 hours.

3. In OPERATIONAL MODE 5[a] with a withdrawn control rod not coupled to its associated drive mechanism, within 2 hours:

   a. Insert the control rod to accomplish recoupling and verify recoupling by withdrawing control rod and demonstrating that the control rod will not go to the overtravel position, or

   b. If recoupling is not accomplished, declare the control rod inoperable, fully insert the control rod and disarm the associated directional control valves[b] within one hour, either:

      1) Electrically, or

      2) Hydraulically by closing the drive water and exhaust water isolation valves.

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a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rod and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

b May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

I. Control Rod Position Indication System

All control rod position indicators shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 5[a].

ACTION:

1. In OPERATIONAL MODE 1 or 2 with one or more control rod position indicators inoperable, within one hour either:
   a. Determine the position of the control rod by an alternate method, or
   b. Move the control rod to a position with an OPERABLE position indicator, or
   c. Declare the control rod inoperable, fully insert the inoperable withdrawn control rod(s), and disarm the associated directional control valves[b] either:
      1) Electrically, or
      2) Hydraulically by closing the drive water and exhaust water isolation valves.

4.3 - SURVEILLANCE REQUIREMENTS

I. Control Rod Position Indication System

The control rod position indication system shall be determined OPERABLE by verifying:

1. At least once per 24 hours that the position of each control rod is indicated.
2. That the indicated control rod position changes during the movement of the control rod drive when performing Surveillance Requirement 4.3.C.1.
3. That the control rod position indicator corresponds to the control rod position indicated by the "Full out" position indicator when performing Surveillance Requirement 4.3.H.2.

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a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

b May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod(s) to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

2. With the provisions of ACTION 1 above not met, be in at least HOT SHUTDOWN within the next 12 hours.

3. In OPERATIONAL MODE 5[a] with a withdrawn control rod position indicator inoperable:

   a. Move the control rod to a position with an OPERABLE position indicator, or

   b. Fully insert the control rod.

4.3 - SURVEILLANCE REQUIREMENTS

a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

3.3 - LIMITING CONDITIONS FOR OPERATION

J. Control Rod Drive Housing Support

The control rod drive housing support shall be in place.

APPLICABILITY:

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

ACTION:

With the control rod drive housing support not in place, be in at least HOT SHUTDOWN within 12 hours and in at least COLD SHUTDOWN within the following 24 hours.

4.3 - SURVEILLANCE REQUIREMENTS

J. Control Rod Drive Housing Support

The control rod drive housing support shall be verified to be in place by a visual inspection prior to startup any time it has been disassembled or when maintenance has been performed in the control rod drive housing support area.

3.3 - LIMITING CONDITIONS FOR OPERATION

K. SDV Vent and Drain Valves

All scram discharge volume (SDV) vent and drain valves shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

1. With[b] one or more SDV vent or drain lines with one valve inoperable, isolate[c] the associated line within 7 days or be in HOT SHUTDOWN within the next 12 hours.
2. With[b] one or more SDV vent or drain lines with both valves inoperable, isolate[c] the associated line within 8 hours or be in HOT SHUTDOWN within the next 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

K. SDV Vent and Drain Valves

The scram discharge volume vent and drain valves shall be demonstrated OPERABLE:

1. At least once per 31 days by verifying each valve to be open[a], and
2. At least once per 92 days by cycling each valve through at least one complete cycle of travel.
3. At least once per 18 months, the scram discharge volume vent and drain valves shall be demonstrated to:
   a. Close within 30 seconds after receipt of a signal for control rods to scram, and
   b. Open after the scram signal is reset.

b Separate Action statement entry is allowed for each SDV vent and drain line.

c An isolated line may be unisolated under administrative control to allow draining and venting of the SDV.

a These valves may be closed intermittently for testing under administrative controls.

3.3 - LIMITING CONDITIONS FOR OPERATION

L. Rod Worth Minimizer (RWM)

The rod worth minimizer (RWM) shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2[a], when THERMAL POWER is less than or equal to 20% of RATED THERMAL POWER.

ACTION:

With the RWM inoperable, verify control rod movement and compliance with the prescribed control rod pattern by a second licensed operator or technically qualified individual who is present at the reactor control console. Otherwise, control rod movement may be made only by actuating the manual scram or placing the reactor mode switch in the Shutdown position.

4.3 - SURVEILLANCE REQUIREMENTS

L. Rod Worth Minimizer (RWM)

The RWM shall be demonstrated OPERABLE:

1. By verifying that the control rod patterns and sequence input to the RWM computer are correctly loaded following any loading of the program into the computer.

2. In OPERATIONAL MODE 2 within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical:

   a. by verifying proper indication of the selection error of at least one out-of-sequence control rod.

   b. by verifying the rod block function by demonstrating inability to withdraw an out-of-sequence control rod.

3. In OPERATIONAL MODE 1 prior to reducing THERMAL POWER below 20% of RATED THERMAL POWER:

   a. by verifying proper indication of the selection error of at least one out-of-sequence control rod.

   b. by verifying the rod block function by demonstrating inability to withdraw an out-of-sequence control rod.

---

a Entry into OPERATIONAL MODE 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RWM prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

3.3 - LIMITING CONDITIONS FOR OPERATION

M. Rod Block Monitor (RBM)

Both rod block monitor (RBM) CHANNEL(s) shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE 1, when thermal power is greater than or equal to 30% of RATED THERMAL POWER.

ACTION:

1. With one RBM CHANNEL inoperable:
   a. Verify that the reactor is not operating in a LIMITING CONTROL ROD PATTERN, and
   b. Restore the inoperable RBM CHANNEL to OPERABLE status within 24 hours.
2. With the provisions of ACTION 1 above not met, place the inoperable rod block monitor CHANNEL in the tripped condition within the next one hour.
3. With both RBM CHANNEL(s) inoperable, place at least one inoperable rod block monitor CHANNEL in the tripped condition within one hour.

4.3 - SURVEILLANCE REQUIREMENTS

M. Rod Block Monitor (RBM)

Each of the required RBM CHANNEL(s) shall be demonstrated OPERABLE by performance of a:

1. CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies and for the OPERATIONAL MODE(s) specified in Table 4.2.E-1.
2. CHANNEL FUNCTIONAL TEST prior to control rod withdrawal when the reactor is operating in a LIMITING CONTROL ROD PATTERN, but no more often than daily.

3.3 - LIMITING CONDITIONS FOR OPERATION

N. Economic Generation Control (EGC) System

The economic generation control (EGC) system may be in operation with automatic flow control provided:

a. Core flow is within 65% to 100% of rated core flow, and

b. THERMAL POWER is ≥20% of RATED THERMAL POWER.

APPLICABILITY

OPERATIONAL MODE 1.

ACTION:

With core flow less than 65% or greater than 100% of rated core flow, or THERMAL POWER less than 20% of RATED THERMAL POWER, restore operation to within the limits within one hour. Otherwise, immediately remove the plant from EGC operation.

4.3 - SURVEILLANCE REQUIREMENTS

N. Economic Generation Control (EGC) System

The economic generation control system shall be demonstrated OPERABLE by verifying that core flow is within 65% to 100% of rated core flow and THERMAL POWER is ≥20% of RATED THERMAL POWER:

a. Prior to entry into EGC operation, and

b. At least once per 12 hours while operating in EGC.

BASES

3/4.3.A SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The SHUTDOWN MARGIN limitation is a restriction to be applied principally to a new refueling pattern. Satisfaction of the limitation must be determined at the time of loading and must be such that it will apply to the entire subsequent fuel cycle. This determination is provided by core design calculations and administrative control of fuel loading patterns. These procedures include restrictions to allow only those intermediate fuel assembly configurations that have been shown to provide the required SHUTDOWN MARGIN.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed during xenon free conditions and adjusted to 68°F to accommodate the current moderator temperature. The generalized form is that the reactivity of the core loading will be limited so the core can be made subcritical by at least R + 0.35% Δk/k or R + 0.25% Δk/k, as appropriate, with the strongest control rod fully withdrawn and all others fully inserted. Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of determination of the highest control rod worth, either analytically or by test. This is due to the reduced uncertainty in the SHUTDOWN MARGIN test when the highest worth control rod is determined by demonstration. When SHUTDOWN MARGIN is determined by calculations not associated with a test, additional margin must be added to the specified SHUTDOWN MARGIN limit to account for uncertainties in the calculation.

The value of R in units of % Δk/k is the difference between the calculated beginning-of-life core reactivity, at the beginning of the operating cycle, and the calculated value of maximum core reactivity at any time later in the operating cycle, where it would be greater than at the beginning. The value of R shall include the potential SHUTDOWN MARGIN loss assuming full B_4C settling in all inverted poison tubes present in the core. R must be a positive quantity or zero and a new value of R must be determined for each new fuel cycle.

The value of % Δk/k in the above expression is provided as a finite, demonstrable, subcriticality margin. This margin is verified using an in-sequence control rod withdrawal at the beginning-of-life fuel cycle conditions. This assures subcriticality with not only the strongest fully withdrawn but at least an R + 0.25% Δk margin beyond this condition. This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined anytime a control rod is incapable of insertion following a scram signal. Any control rod that is immovable as a result of excessive friction or mechanical interference, or is known to be unscrammable, per Specification 3.3.C, is considered to be incapable of insertion following a scram signal. It is important to note that a control rod can be electrically immovable, but scrammable, and no increase in SHUTDOWN MARGIN is required for these control rods.

BASES

3/4.3.B Reactivity Anomalies

During each fuel cycle, excess operating reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of the critical rod pattern selected base states to the predicted rod inventory at that state. Power operating base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons. Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1% $\Delta k/k$. Deviations in core reactivity greater than 1% $\Delta k/k$ are not expected and require thorough evaluation. A 1% $\Delta k/k$ reactivity limit is considered safe since an insertion of the reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

3/4.3.C Control Rod OPERABILITY

Control rods are the primary reactivity control system for the reactor. In conjunction with the Reactor Protection System, the control rods provide the means for reliable control of reactivity changes to ensure the specified acceptable fuel design limits are not exceeded. This specification, along with others, assures that the performance of the control rods in the event of an accident or transient, meets the assumptions used in the safety analysis. Of primary concern is the trippability of the control rods. Other causes for inoperability are addressed in other Specifications following this one. However, the inability to move a control rod which remains trippable does not prevent the performance of the control rod's safety function.

The specification requires that a rod be taken out-of-service if it cannot be moved with drive pressure. Damage within the control rod drive mechanism could be a generic problem, therefore with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods. Control rods that are inoperable due to exceeding allowed scram times, but are movable by control rod drive pressure, need not be disarmed electrically if the shutdown margin provisions are met for each position of the affected rod(s).

If the rod is fully inserted and then disarmed electrically or hydraulically, it is in a safe position of maximum contribution to shutdown reactivity. (Note: To disarm the drive electrically, four amphenol-type plug connectors are removed from the drive insert and withdrawal solenoids, rendering the drive immovable. This procedure is equivalent to valving out the drive and is preferred, as drive water cools and minimizes crud accumulation in the drive.). If it is disarmed electrically in a non-fully inserted position, that position shall be consistent with the SHUTDOWN MARGIN limitation stated in Specification 3.3.A. This assures that the core can be shut down at all times with the remaining control rods, assuming the strongest OPERABLE control rod does not

BASES

insert. The occurrence of more than eight inoperable control rods could be indicative of a generic control rod drive problem which requires prompt investigation and resolution.

In order to reduce the potential for Control Rod Drive (CRD) damage and more specifically, collet housing failure, a program of disassembly and inspection of CRDs is conducted during or after each refueling outage. This program follows the recommendations of General Electric SIL-139 with nondestructive examination results compiled and reported to General Electric on collet housing cracking problems.

The required surveillance intervals are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

<u>3/4.3.D</u> <u>Control Rod Maximum Scram Insertion Times;</u>

<u>3/4.3.E</u> <u>Control Rod Average Scram Insertion Times; and</u>

<u>3/4.3.F</u> <u>Four Control Rod Group Scram Insertion Times</u>

These specifications ensure that the control rod insertion times are consistent with those used in the safety analyses. The control rod system is analyzed to bring the reactor subcritical at a rate fast enough to prevent fuel damage, i.e., to prevent the MCPR from becoming less than the fuel cladding integrity Safety Limit. The analyses demonstrate that if the reactor is operated within the limitation set in Specification 3.11.C, the negative reactivity insertion rates associated with the scram performance (as adjusted for statistical variation in the observed data) result in protection of the MCPR Safety Limit.

Analysis of the limiting power transient shows that the negative reactivity rates, resulting from the scram with the average response of all the drives, as given in the above specification, provide the required protection, and MCPR remains greater than the fuel cladding integrity SAFETY LIMIT. In the analytical treatment of most transients, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typically observed time delay of about 210 milliseconds. Approximately 90 milliseconds after neutron flux reaches the trip point, the pilot scram valve solenoid de-energizes and 120 milliseconds later the control rod motion is estimated to actually begin. However, 200 milliseconds rather than 120 milliseconds is conservatively assumed for this time interval in the transient analyses and is also included in the allowable scram insertion times specified in Specifications 3.3.D, 3.3.E, and 3.3.F. In the statistical treatment of the limiting transients, a statistical distribution of total scram delay is used rather than the bounding value described above.

BASES

The performance of the individual control rod drives is monitored to assure that scram performance is not degraded. Observed plant data or Technical Specification limits were used to determine the average scram performance used in the transient analyses, and the results of each set of control rod scram tests performed during the current cycle are compared against earlier results to verify that the performance of the control rod insertion system has not changed significantly.

If test results should be determined to fall outside of the statistical population defining the scram performance characteristics used in the transient analyses, a re-determination of thermal margin requirements is undertaken as required by Specification 3.11.C. A smaller test sample than that required by these specifications is not statistically significant and should not be used in the re-determination of thermal margins. Individual control rod drives with excessive scram times can be fully inserted into the core and de-energized in the manner of an inoperable rod drive provided the allowable number of inoperable control rod drives is not exceeded. In this case, the scram speed of the drive shall not be used as a basis in the re-determination of thermal margin requirements. For excessive average scram insertion times, only the individual control rods in the two-by-two array which exceed the allowed average scram insertion time are considered inoperable.

The scram times for all control rods are measured at the time of each refueling outage. Experience with the plant has shown that control drive insertion times vary little through the operating cycle; hence no re-assessment of thermal margin requirements is expected under normal conditions. The history of drive performance accumulated to date indicates that the 90% insertion times of new and overhauled drives approximate a normal distribution about the mean which tends to become skewed toward longer scram times as operating time is accumulated. The probability of a drive not exceeding the mean 90% insertion time by 0.75 seconds is greater than 0.999 for a normal distribution. The measurement of the scram performance of the drives surrounding a drive, which exceeds the expected range of scram performance, will detect local variations and also provide assurance that local scram time limits are not exceeded. Continued monitoring of other drives exceeding the expected range of scram times provides surveillance of possible anomalous performance.

The test schedule provides reasonable assurance of detection of slow drives before system deterioration beyond the limits of Specification 3.3.C. The program was developed on the basis of the statistical approach outlined above and judgement. The occurrence of scram times within the limits, but significantly longer than average, should be viewed as an indication of a systematic problem with control rod drives, especially if the number of drives exhibiting such scram times exceeds eight, which is the allowable number of inoperable rods.

3/4.3.G Control Rod Scram Accumulators

The control rod scram accumulators are part of the control rod drive system and are provided to ensure that the control rods scram under varying reactor conditions. The control rod scram accumulators store sufficient energy to fully insert a control rod at any reactor vessel pressure. The accumulator is a hydraulic cylinder with a free floating piston. The piston separates the water

BASES

used to scram the control rods from the nitrogen which provides the required energy. The scram accumulators are necessary to scram the control rods within the required insertion times.

Control rods with inoperable accumulators are declared inoperable and Specification 3.3.C then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram than has been analyzed even though control rods with inoperable accumulators may still be inserted with normal drive water pressure. OPERABILITY of the accumulator ensures that there is a means available to insert the control rods even under the most unfavorable depressurization of the reactor.

3/4.3.H Control Rod Drive Coupling

Control rod dropout accidents can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod drop accident is eliminated. Neutron instrumentation response to rod movement may provide verification that a rod is following its drive. Absence of such response to drive movement may indicate an uncoupled condition, or may be due to the lack of proximity of the drive to the instrumentation. However, the overtravel position feature provides a positive check, as only uncoupled drives may reach this position.

3/4.3.I Control Rod Position Indication System (RPIS)

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE. Normal control rod position is displayed by two-digit indication to the operator from position 00 to 48. Each even number is a latching position, whereas each odd number provides information while the rod is in-motion and inputs for rod drift annunciation. The ACTION statement provides for the condition where no positive information is displayed for a large portion or all of the rod's travel. Usually, only one digit of one or two of a rod's positions is unavailable with a faulty RPIS, and the control rod may be located in a known position. However, there are several alternate methods for determining control rod position including the full core display, the four rod display, the rod worth minimizer, and the process computer. Additionally, there are independent "full-in" and "full-out" indicators at the 00 and 48 positions. Another method to determine position would be to move the control rod, by single notch movement, to a position with an OPERABLE position indicator. The original position would then be established and the control rod could be returned to its original position by single notch movement. As long as no control rod drift alarms are received, the position of the control rod would then be known.

3/4.3.J Control Rod Drive Housing Support

The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in Section 4.6.3.5 of the UFSAR. This support is not required if the reactor coolant system is at atmospheric pressure, since there would then be no driving force to rapidly eject a drive housing.

3/4.3.K Scram Discharge Volume Vent and Drain Valves

The scram discharge volume is required to be OPERABLE so that it will be available when needed to accept discharge water from the control rods during a reactor scram and will isolate the reactor coolant system from the containment when required. The operability of the scram discharge volume vent and drain valves assures the proper venting and draining of the volume, so that water accumulation in the volume does not occur. These specifications designate the minimum acceptable level of scram discharge volume vent and drain valve OPERABILITY, provide for the periodic verification that the valves are open, and for the testing of these valves under reactor scram conditions during each refueling outage.

3/4.3.L Rod Worth Minimizer

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. These sequences are developed prior to initial operation of the unit following any refueling outage and the requirement that an operator follow these sequences is supervised by the RWM or a second technically qualified individual. These sequences are developed to limit reactivity worth of control rods and, together with the integral rod velocity limiters and the action of the control rod drive system, limit potential reactivity insertion such that the results of a control rod drop accident will not exceed a maximum fuel energy content of 280 cal/gm. The peak fuel enthalpy of 280 cal/gm is below the energy content at which rapid fuel dispersal and primary system damage have been found to occur based on experimental data. Therefore, the energy deposited during a postulated rod drop accident is significantly less than that required for rapid fuel dispersal.

The analysis of the control rod drop accident was originally presented in Sections 7.9.3, 14.2.1.2, and 14.2.1.4 of the original SAR. Improvements in analytical capability have allowed a more refined analysis of the control rod drop accident which is discussed below.

Every operating cycle the peak fuel rod enthalpy rise is determined by comparing cycle specific parameters with the results of parametric analyses. This peak fuel rod enthalpy is then compared to the analysis limit of 280 cal/gm to demonstrate compliance for that operating cycle. If the cycle specific parameters are outside the range used in the parametric study, an extension of the enthalpy may be required. Some of the cycle specific parameters used in the analysis are: maximum control rod worth, Doppler coefficient, effective delayed neutron fraction and maximum

BASES

four bundle local peaking factor. The NRC approved methodology listed in Specification 6.6.A.4 provides a detailed description of the methodology used in performing the rod drop analyses.

The rod worth minimizer provides automatic supervision to assure that out-of-sequence control rods will not be withdrawn or inserted, i.e., it limits operator deviations from planned withdrawal sequences (reference UFSAR Section 7.7.2). It serves as a backup to procedural control of control rod worth. In the event that the rod worth minimizer is out-of-service when required, a second licensed operator or other technically qualified individual who is present at the reactor console can manually fulfill the control rod pattern conformance function of the rod worth minimizer. In this case, the normal procedural controls are backed up by independent procedural controls to assure conformance.

3/4.3.M Rod Block Monitor

The rod block monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator, who withdraws control rods according to a written sequence. The specified restrictions with one channel out-of-service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

3/4.3.N Economic Generation Control System

Operation of the facility with the economic generation control system (EGC) (automatic flow control) is limited to the range of 65% to 100% of rated core flow. In this flow range and above 20% of RATED THERMAL POWER, the reactor could safely tolerate a rate of change of load of 8 MWe/sec (reference UFSAR Section 7.7.3.2). Limits within the EGC and the flow control system prevent rates of change greater than approximately 4 MWe/sec. When EGC is in operation, this fact will be indicated on the main control room console.

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

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-254

QUAD CITIES NUCLEAR POWER STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 158
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 July 29, 1992, as supplemented by letters dated January 14, 1993, February 16, 1993, and May 9, 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. 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: July 27, 1995

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

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-265

QUAD CITIES NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 154
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 July 29, 1992, as supplemented by letters dated January 14, 1993, February 16, 1993, and May 9, 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. 154 , 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: July 27, 1995

ATTACHMENT TO LICENSE AMENDMENT NOS. 158 AND 154

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
3.3/4.3-1	3.3/4.3-1	3/4.3-1
3.3/4.3-2	3.3/4.3-1a	3/4.3-2
3.3/4.3-3	3.3/4.3-2	3/4.3-3
3.3/4.3-4	3.3/4.3-3	3/4.3-4
3.3/4.3-5	3.3/4.3-4	3/4.3-5
3.3/4.3-6	3.3/4.3-5	3/4.3-6
3.3/4.3-7	3.3/4.3-6	3/4.3-7
3.3/4.3-8	---	3/4.3-8
---	---	3/4.3-9
---	---	3/4.3-10
---	---	3/4.3-11
---	---	3/4.3-12
---	---	3/4.3-13
---	---	3/4.3-14
---	---	3/4.3-15
---	---	3/4.3-16
---	---	3/4.3-17
---	---	3/4.3-18
---	---	3/4.3-19
---	---	3/4.3-20
3.3/4.3-9	3.3/4.3-7	B 3/4.3-1
3.3/4.3-10	3.3/4.3-7a	B 3/4.3-2
3.3/4.3-11	3.3/4.3-8	B 3/4.3-3
3.3/4.3-12	3.3/4.3-9	B 3/4.3-4
3.3/4.3-13	3.3/4.3-10	B 3/4.3-5
3.3/4.3-14	3.3/4.3-11	B 3/4.3-6
3.3/4.3-15	---	B 3/4.3-7
3.3/4.3-16	---	---
3.3/4.3-17	---	---

3.3 - LIMITING CONDITIONS FOR OPERATION

A. SHUTDOWN MARGIN (SDM)

The SHUTDOWN MARGIN (SDM) shall be equal to or greater than:

1. 0.35% Δk/k with the highest worth control rod analytically determined, or
2. 0.25% Δk/k with the highest worth control rod determined by test.

APPLICABILITY:

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

ACTION:

With the SHUTDOWN MARGIN less than specified:

1. In OPERATIONAL MODE 1 or 2, restore the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.
2. In OPERATIONAL MODE 3 or 4, immediately verify all insertable control rods to be fully inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL MODE 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.
3. In OPERATIONAL MODE 5, suspend CORE ALTERATION(s) and other activities that could reduce the SHUTDOWN MARGIN and fully insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

4.3 - SURVEILLANCE REQUIREMENTS

A. SHUTDOWN MARGIN

The SHUTDOWN MARGIN shall be determined to be equal to or greater than that specified at any time during the operating cycle:

1. By demonstration, prior to or during the first startup after each refueling outage.
2. Within 24 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or known to be unscrammable. The required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or unscrammable control rod.
3. By calculation, prior to each fuel movement during the fuel loading sequence.

3.3 - LIMITING CONDITIONS FOR OPERATION

B. Reactivity Anomalies

The reactivity equivalence of the difference between the actual ROD DENSITY and the predicted ROD DENSITY shall not exceed 1% Δk/k.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

With the reactivity equivalence difference exceeding 1% Δk/k, within 12 hours perform an analysis to determine and explain the cause of the reactivity difference; operation may continue if the difference is explained and corrected.

With the provisions of the ACTION above not met, be in at least HOT SHUTDOWN within the next 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

B. Reactivity Anomalies

The reactivity equivalence of the difference between the actual ROD DENSITY and the predicted ROD DENSITY shall be verified to be less than or equal to 1% Δk/k:

1. During the first startup following CORE ALTERATION(s), and
2. At least once per 31 effective full power days.

3.3 - LIMITING CONDITIONS FOR OPERATION

C. Control Rod OPERABILITY

All control rods shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

1. With one control rod inoperable due to being immovable as a result of excessive friction or mechanical interference, or known to be unscrammable:

   a. Within one hour:

      1) Verify that the inoperable control rod, if withdrawn, is separated from all other inoperable withdrawn control rods by at least two control cells in all directions.

      2) Disarm the associated directional control valves[a] either:

         a) Electrically, or

         b) Hydraulically by closing the drive water and exhaust water isolation valves.

   b. With the provisions of ACTION 1.a above not met, be in at least HOT SHUTDOWN within the next 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

C. Control Rod OPERABILITY

1. When above the low power setpoint of the RWM, all withdrawn control rods not required to have their directional control valves disarmed electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch:

   a. At least once per 7 days, and

   b. At least once per 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference, or known to be unscrammable.

2. All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.3.D, 4.3.F, 4.3.G, 4.3.H and 4.3.I.

---

a May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

4.3 - SURVEILLANCE REQUIREMENTS

c. Comply with Surveillance Requirement 4.3.A.2 within 24 hours or be in HOT SHUTDOWN within the next 12 hours.

2. With one or more control rods scrammable but inoperable for causes other than addressed in ACTION 3.3.C.1 above:

   a. If the inoperable control rod(s) is withdrawn, within one hour:

      1) Verify that the inoperable withdrawn control rod(s) is separated from all other inoperable withdrawn control rods by at least two control cells in all directions, and

      2) Demonstrate the insertion capability of the inoperable withdrawn control rod(s) by inserting the inoperable withdrawn control rod(s) at least one notch by drive water pressure within the normal operating range.[b]

   b. With the provisions of ACTION 2.a above not met, fully insert the inoperable withdrawn control rod(s) and disarm the associated directional control valves[a] either:

      1) Electrically, or

      2) Hydraulically by closing the drive water and exhaust water isolation valves.

---

b The inoperable control rod may then be withdrawn to a position no further withdrawn than its position when found to be inoperable.

a May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

c. If the inoperable control rod(s) is fully inserted, within one hour disarm the associated directional control valves[a] either:

1) Electrically, or

2) Hydraulically by closing the drive water and exhaust water isolation valves.

3. With the provisions of ACTION 2 above not met, be in at least HOT SHUTDOWN within the next 12 hours.

4. With more than 8 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

a May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

D. Maximum Scram Insertion Times

The maximum scram insertion time of each control rod from the fully withdrawn position to 90% insertion, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7 seconds.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding 7 seconds:

1. Declare the control rod(s) exceeding the above maximum scram insertion time inoperable, and
2. When operation is continued with three or more control rods with maximum scram insertion times in excess of 7 seconds, perform Surveillance Requirement 4.3.D.3 at least once per 60 days of POWER OPERATION.

With the provisions of the ACTION above not met, be in at least HOT SHUTDOWN within 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

D. Maximum Scram Insertion Times

The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than 800 psig and, during single control rod scram time tests, with the control rod drive pumps isolated from the accumulators:

1. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER:
   a. following CORE ALTERATION(s), or
   b. after a reactor shutdown that is greater than 120 days,
2. For specifically affected individual control rods[a] following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and
3. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of POWER OPERATION.

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a The provisions of Specification 4.0.D are not applicable provided this surveillance is conducted prior to exceeding 40% of RATED THERMAL POWER.

3.3 - LIMITING CONDITIONS FOR OPERATION

E. Average Scram Insertion Times

The average scram insertion time of all OPERABLE control rods from the fully withdrawn position, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.900
50	2.00
90	3.50

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

With the average scram insertion time exceeding any of the above limits, be in at least HOT SHUTDOWN within 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

E. Average Scram Insertion Times

The control rod average scram times shall be demonstrated by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.3.D.

3.3 - LIMITING CONDITIONS FOR OPERATION

F. Group Scram Insertion Times

The average of the scram insertion times, from the fully withdrawn position, for the three fastest control rods of all groups of four control rods in a two-by-two array, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed any of the following:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.120
90	3.800

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

With the average scram insertion times of control rods exceeding the above limits:

1. Declare the control rods exceeding the above average scram insertion times inoperable until an analysis is performed to determine that required scram reactivity remains for the slow four control rod group, and

2. When operation is continued with an average scram insertion time(s) in excess of the average scram insertion time limit, perform Surveillance Requirement 4.3.D.3 at least once per 60 days of POWER OPERATION.

With the provisions of the ACTION(s) above not met, be in at least HOT SHUTDOWN within 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

F. Group Scram Insertion Times

All control rods shall be demonstrated OPERABLE by scram time testing from the fully withdrawn position as required by Surveillance Requirement 4.3.D.

3.3 - LIMITING CONDITIONS FOR OPERATION

G. Control Rod Scram Accumulators

All control rod scram accumulators shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 5[a].

ACTION:

1. In OPERATIONAL MODE 1 or 2:
   a. With one control rod scram accumulator inoperable, within 8 hours:
      1) Restore the inoperable accumulator to OPERABLE status, or
      2) Declare the control rod associated with the inoperable accumulator inoperable.
   b. With the provisions of ACTION 1.a above not met, be in at least HOT SHUTDOWN within the next 12 hours.
   c. With more than one control rod scram accumulator inoperable, declare the associated control rods inoperable and:

4.3 - SURVEILLANCE REQUIREMENTS

G. Control Rod Scram Accumulators

Each control rod scram accumulator shall be determined OPERABLE at least once per 7 days by verifying that the indicated pressure is ≥800 psig unless the control rod is fully inserted and disarmed, or scrammed.

a In OPERATIONAL MODE 5, this Specification is applicable for the accumulators associated with each withdrawn control rod and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

3.3 - LIMITING CONDITIONS FOR OPERATION

1) If the control rod associated with any inoperable scram accumulator is withdrawn, immediately verify that at least one control rod drive pump is operating by inserting at least one withdrawn control rod at least one notch. With no control rod drive pump operating, immediately place the reactor mode switch in the Shutdown position.

2) Fully insert the inoperable control rods and disarm the associated directional control valves[b] either:

a) Electrically, or

b) Hydraulically by closing the drive water and exhaust water isolation valves.

d. With the provisions of ACTION 1.c.2 above not met, be in at least HOT SHUTDOWN within 12 hours.

2. In OPERATIONAL MODE 5[a]

a. With one withdrawn control rod with its associated scram accumulator inoperable, fully insert the affected control rod and disarm the associated directional control valves[b] within one hour, either:

4.3 - SURVEILLANCE REQUIREMENTS

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a In OPERATIONAL MODE 5, this Specification is applicable for the accumulators associated with each withdrawn control rod and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

b May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

1) Electrically, or

2) Hydraulically by closing the drive water and exhaust water isolation valves.

b. With more than one withdrawn control rod with the associated scram accumulator inoperable or no control rod drive pump operating, immediately place the reactor mode switch in the Shutdown position.

4.3 - SURVEILLANCE REQUIREMENTS

3.3 - LIMITING CONDITIONS FOR OPERATION

H. Control Rod Drive Coupling

All control rods shall be coupled to their drive mechanisms.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 5[a].

ACTION:

1. In OPERATIONAL MODE 1 or 2 with one control rod not coupled to its associated drive mechanism, within 2 hours:

   a. If permitted by the RWM, insert the control rod drive mechanism to accomplish recoupling and verify recoupling by withdrawing the control rod, and:

      1) Observing any indicated response of the nuclear instrumentation, and

      2) Demonstrating that the control rod will not go to the overtravel position.

   b. If not permitted by the RWM or, if recoupling is not accomplished in accordance with ACTION 1.a above, then declare the control rod inoperable, fully insert the control rod and disarm the associated directional control valves[b] either:

      1) Electrically, or

4.3 - SURVEILLANCE REQUIREMENTS

H. Control Rod Drive Coupling

Each affected control rod shall be demonstrated to be coupled to its drive mechanism by observing any indicated response of the nuclear instrumentation while withdrawing the control rod to the fully withdrawn position and then verifying that the control rod drive does not go to the overtravel position:

1. Prior to reactor criticality after completing CORE ALTERATION(s) that could have affected the control rod drive coupling integrity,

2. Anytime the control rod is withdrawn to the "Full out" position in subsequent operation, and

3. Following maintenance on or modification to the control rod or control rod drive system which could have affected the control rod drive coupling integrity.

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a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

b May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

2) Hydraulically by closing the drive water and exhaust water isolation valves.

2. With the provisions of ACTION 1 above not met, be in at least HOT SHUTDOWN within 12 hours.

3. In OPERATIONAL MODE 5[a] with a withdrawn control rod not coupled to its associated drive mechanism, within 2 hours:

    a. Insert the control rod to accomplish recoupling and verify recoupling by withdrawing control rod and demonstrating that the control rod will not go to the overtravel position, or

    b. If recoupling is not accomplished, declare the control rod inoperable, fully insert the control rod and disarm the associated directional control valves[b] within one hour, either:

        1) Electrically, or

        2) Hydraulically by closing the drive water and exhaust water isolation valves.

4.3 - SURVEILLANCE REQUIREMENTS

a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

b May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

I. Control Rod Position Indication System

All control rod position indicators shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 5[a].

ACTION:

1. In OPERATIONAL MODE 1 or 2 with one or more control rod position indicators inoperable, within one hour either:
   a. Determine the position of the control rod by an alternate method, or
   b. Move the control rod to a position with an OPERABLE position indicator, or
   c. Declare the control rod inoperable, fully insert the inoperable withdrawn control rod(s), and disarm the associated directional control valves[b] either:
      1) Electrically, or
      2) Hydraulically by closing the drive water and exhaust water isolation valves.

4.3 - SURVEILLANCE REQUIREMENTS

I. Control Rod Position Indication System

The control rod position indication system shall be determined OPERABLE by verifying:

1. At least once per 24 hours that the position of each control rod is indicated.
2. That the indicated control rod position changes during the movement of the control rod drive when performing Surveillance Requirement 4.3.C.1.
3. That the control rod position indicator corresponds to the control rod position indicated by the "Full out" position indicator when performing Surveillance Requirement 4.3.H.2.

a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

b May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod(s) to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

2. With the provisions of ACTION 1 above not met, be in at least HOT SHUTDOWN within the next 12 hours.

3. In OPERATIONAL MODE 5[a] with a withdrawn control rod position indicator inoperable:

   a. Move the control rod to a position with an OPERABLE position indicator, or

   b. Fully insert the control rod.

4.3 - SURVEILLANCE REQUIREMENTS

a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

3.3 - LIMITING CONDITIONS FOR OPERATION

J. Control Rod Drive Housing Support

The control rod drive housing support shall be in place.

APPLICABILITY:

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

ACTION:

With the control rod drive housing support not in place, be in at least HOT SHUTDOWN within 12 hours and in at least COLD SHUTDOWN within the following 24 hours.

4.3 - SURVEILLANCE REQUIREMENTS

J. Control Rod Drive Housing Support

The control rod drive housing support shall be verified to be in place by a visual inspection prior to startup any time it has been disassembled or when maintenance has been performed in the control rod drive housing support area.

3.3 - LIMITING CONDITIONS FOR OPERATION

K. SDV Vent and Drain Valves

All scram discharge volume (SDV) vent and drain valves shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

1. With[b] one or more SDV vent or drain lines with one valve inoperable, isolate[c] the associated line within 7 days or be in HOT SHUTDOWN within the next 12 hours.

2. With[b] one or more SDV vent or drain lines with both valves inoperable, isolate[c] the associated line within 8 hours or be in HOT SHUTDOWN within the next 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

K. SDV Vent and Drain Valves

The scram discharge volume vent and drain valves shall be demonstrated OPERABLE:

1. At least once per 31 days by verifying each valve to be open[a], and

2. At least once per 92 days by cycling each valve through at least one complete cycle of travel.

3. At least once per 18 months, the scram discharge volume vent and drain valves shall be demonstrated to:

   a. Close within 30 seconds after receipt of a signal for control rods to scram, and

   b. Open after the scram signal is reset.

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b Separate Action statement entry is allowed for each SDV vent and drain line.

c An isolated line may be unisolated under administrative control to allow draining and venting of the SDV.

a These valves may be closed intermittently for testing under administrative controls.

3.3 - LIMITING CONDITIONS FOR OPERATION

L. Rod Worth Minimizer (RWM)

The rod worth minimizer (RWM) shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2[a], when thermal power is less than or equal to 10% of RATED THERMAL POWER.

ACTION:

With the RWM inoperable, verify control rod movement and compliance with the prescribed control rod pattern by a second licensed operator or technically qualified individual who is present at the reactor control console. Otherwise, control rod movement may be made only by actuating the manual scram or placing the reactor mode switch in the Shutdown position.

4.3 - SURVEILLANCE REQUIREMENTS

L. Rod Worth Minimizer (RWM)

The RWM shall be demonstrated OPERABLE:

1. By verifying that the control rod patterns and sequence input to the RWM computer are correctly loaded following any loading of the program into the computer.

2. In OPERATIONAL MODE 2 within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical:

   a. by verifying proper indication of the selection error of at least one out-of-sequence control rod.

   b. by verifying the rod block function by demonstrating inability to withdraw an out-of-sequence control rod.

3. In OPERATIONAL MODE 1 prior to reducing thermal power below 10% of RATED THERMAL POWER:

   a. by verifying proper indication of the selection error of at least one out-of-sequence control rod.

   b. by verifying the rod block function by demonstrating inability to withdraw an out-of-sequence control rod.

a Entry into OPERATIONAL MODE 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RWM prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

3.3 - LIMITING CONDITIONS FOR OPERATION

M. Rod Block Monitor (RBM)

Both rod block monitor (RBM) CHANNEL(s) shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE 1, when thermal power is greater than or equal to 30% of RATED THERMAL POWER.

ACTION:

1. With one RBM CHANNEL inoperable:
   a. Verify that the reactor is not operating in a LIMITING CONTROL ROD PATTERN, and
   b. Restore the inoperable RBM CHANNEL to OPERABLE status within 24 hours.
2. With the provisions of ACTION 1 above not met, place the inoperable rod block monitor CHANNEL in the tripped condition within the next one hour.
3. With both RBM CHANNEL(s) inoperable, place at least one inoperable rod block monitor CHANNEL in the tripped condition within one hour.

4.3 - SURVEILLANCE REQUIREMENTS

M. Rod Block Monitor (RBM)

Each of the required RBM CHANNEL(s) shall be demonstrated OPERABLE by performance of a:

1. CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies and for the OPERATIONAL MODE(s) specified in Table 4.2.E-1.
2. CHANNEL FUNCTIONAL TEST prior to control rod withdrawal when the reactor is operating in a LIMITING CONTROL ROD PATTERN, but no more often than daily.

3.3 - LIMITING CONDITIONS FOR OPERATION

N. Economic Generation Control (EGC) System

The economic generation control (EGC) system may be in operation with automatic flow control provided:

a. Core flow is within 65% to 100% of rated core flow, and

b. THERMAL POWER is ≥20% of RATED THERMAL POWER.

APPLICABILITY

OPERATIONAL MODE 1.

ACTION:

With core flow less than 65% or greater than 100% of rated core flow, or THERMAL POWER less than 20% of RATED THERMAL POWER, restore operation to within the limits within one hour. Otherwise, immediately remove the plant from EGC operation.

4.3 - SURVEILLANCE REQUIREMENTS

N. Economic Generation Control (EGC) System

The economic generation control system shall be demonstrated OPERABLE by verifying that core flow is within 65% to 100% of rated core flow and THERMAL POWER is ≥20% of RATED THERMAL POWER:

a. Prior to entry into EGC operation, and

b. At least once per 12 hours while operating in EGC.

BASES

3/4.3.A SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The SHUTDOWN MARGIN limitation is a restriction to be applied principally to a new refueling pattern. Satisfaction of the limitation must be determined at the time of loading and must be such that it will apply to the entire subsequent fuel cycle. This determination is provided by core design calculations and administrative control of fuel loading patterns. These procedures include restrictions to allow only those intermediate fuel assembly configurations that have been shown to provide the required SHUTDOWN MARGIN.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed during xenon free conditions and adjusted to 68°F to accommodate the current moderator temperature. The generalized form is that the reactivity of the core loading will be limited so the core can be made subcritical by at least R + 0.35% $\Delta k/k$ or R + 0.25% $\Delta k/k$, as appropriate, with the strongest control rod fully withdrawn and all others fully inserted. Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of determination of the highest control rod worth, either analytically or by test. This is due to the reduced uncertainty in the SHUTDOWN MARGIN test when the highest worth control rod is determined by demonstration. When SHUTDOWN MARGIN is determined by calculations not associated with a test, additional margin must be added to the specified SHUTDOWN MARGIN limit to account for uncertainties in the calculation.

The value of R in units of % $\Delta k/k$ is the difference between the calculated beginning-of-life core reactivity, at the beginning of the operating cycle, and the calculated value of maximum core reactivity at any time later in the operating cycle, where it would be greater than at the beginning. The value of R shall include the potential SHUTDOWN MARGIN loss assuming full B_4C settling in all inverted poison tubes present in the core. R must be a positive quantity or zero and a new value of R must be determined for each new fuel cycle.

The value of % $\Delta k/k$ in the above expression is provided as a finite, demonstrable, subcriticality margin. This margin is verified using an in-sequence control rod withdrawal at the beginning-of-life fuel cycle conditions. This assures subcriticality with not only the strongest fully withdrawn but at least an R + 0.25% Δk margin beyond this condition. This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined anytime a control rod is incapable of insertion following a scram signal. Any control rod that is immovable as a result of excessive friction or mechanical interference, or is known to be unscrammable, per Specification 3.3.C, is considered to be incapable of insertion following a scram signal. It is important to note that a control rod can be electrically immovable, but scrammable, and no increase in SHUTDOWN MARGIN is required for these control rods.

3/4.3.B Reactivity Anomalies

During each fuel cycle, excess operating reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of the critical rod pattern selected base states to the predicted rod inventory at that state. Power operating base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons. Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1% $\Delta k/k$. Deviations in core reactivity greater than 1% $\Delta k/k$ are not expected and require thorough evaluation. A 1% $\Delta k/k$ reactivity limit is considered safe since an insertion of the reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

3/4.3.C Control Rod OPERABILITY

Control rods are the primary reactivity control system for the reactor. In conjunction with the Reactor Protection System, the control rods provide the means for reliable control of reactivity changes to ensure the specified acceptable fuel design limits are not exceeded. This specification, along with others, assures that the performance of the control rods in the event of an accident or transient, meets the assumptions used in the safety analysis. Of primary concern is the trippability of the control rods. Other causes for inoperability are addressed in other Specifications following this one. However, the inability to move a control rod which remains trippable does not prevent the performance of the control rod's safety function.

The specification requires that a rod be taken out-of-service if it cannot be moved with drive pressure. Damage within the control rod drive mechanism could be a generic problem, therefore with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods. Control rods that are inoperable due to exceeding allowed scram times, but are movable by control rod drive pressure, need not be disarmed electrically if the shutdown margin provisions are met for each position of the affected rod(s).

If the rod is fully inserted and then disarmed electrically or hydraulically, it is in a safe position of maximum contribution to shutdown reactivity. (Note: To disarm the drive electrically, four amphenol-type plug connectors are removed from the drive insert and withdrawal solenoids, rendering the drive immovable. This procedure is equivalent to valving out the drive and is preferred, as drive water cools and minimizes crud accumulation in the drive.). If it is disarmed electrically in a non-fully inserted position, that position shall be consistent with the SHUTDOWN MARGIN limitation stated in Specification 3.3.A. This assures that the core can be shut down at all times with the remaining control rods, assuming the strongest OPERABLE control rod does not

BASES

insert. The occurrence of more than eight inoperable control rods could be indicative of a generic control rod drive problem which requires prompt investigation and resolution.

In order to reduce the potential for Control Rod Drive (CRD) damage and more specifically, collet housing failure, a program of disassembly and inspection of CRDs is conducted during or after each refueling outage. This program follows the recommendations of General Electric SIL-139 with nondestructive examination results compiled and reported to General Electric on collet housing cracking problems.

The required surveillance intervals are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

3/4.3.D Control Rod Maximum Scram Insertion Times;

3/4.3.E Control Rod Average Scram Insertion Times; and

3/4.3.F Four Control Rod Group Scram Insertion Times

These specifications ensure that the control rod insertion times are consistent with those used in the safety analyses. The control rod system is analyzed to bring the reactor subcritical at a rate fast enough to prevent fuel damage, i.e., to prevent the MCPR from becoming less than the fuel cladding integrity Safety Limit. The analyses demonstrate that if the reactor is operated within the limitation set in Specification 3.11.C, the negative reactivity insertion rates associated with the scram performance (as adjusted for statistical variation in the observed data) result in protection of the MCPR Safety Limit.

Analysis of the limiting power transient shows that the negative reactivity rates, resulting from the scram with the average response of all the drives, as given in the above specification, provide the required protection, and MCPR remains greater than the fuel cladding integrity SAFETY LIMIT. In the analytical treatment of most transients, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typically observed time delay of about 210 milliseconds. Approximately 90 milliseconds after neutron flux reaches the trip point, the pilot scram valve solenoid de-energizes and 120 milliseconds later the control rod motion is estimated to actually begin. However, 200 milliseconds rather than 120 milliseconds is conservatively assumed for this time interval in the transient analyses and is also included in the allowable scram insertion times specified in Specifications 3.3.D, 3.3.E, and 3.3.F. In the statistical treatment of the limiting transients, a statistical distribution of total scram delay is used rather than the bounding value described above.

BASES

The performance of the individual control rod drives is monitored to assure that scram performance is not degraded. Observed plant data or Technical Specification limits were used to determine the average scram performance used in the transient analyses, and the results of each set of control rod scram tests performed during the current cycle are compared against earlier results to verify that the performance of the control rod insertion system has not changed significantly.

If test results should be determined to fall outside of the statistical population defining the scram performance characteristics used in the transient analyses, a re-determination of thermal margin requirements is undertaken as required by Specification 3.11.C. A smaller test sample than that required by these specifications is not statistically significant and should not be used in the re-determination of thermal margins. Individual control rod drives with excessive scram times can be fully inserted into the core and de-energized in the manner of an inoperable rod drive provided the allowable number of inoperable control rod drives is not exceeded. In this case, the scram speed of the drive shall not be used as a basis in the re-determination of thermal margin requirements. For excessive average scram insertion times, only the individual control rods in the two-by-two array which exceed the allowed average scram insertion time are considered inoperable.

The scram times for all control rods are measured at the time of each refueling outage. Experience with the plant has shown that control drive insertion times vary little through the operating cycle; hence no re-assessment of thermal margin requirements is expected under normal conditions. The history of drive performance accumulated to date indicates that the 90% insertion times of new and overhauled drives approximate a normal distribution about the mean which tends to become skewed toward longer scram times as operating time is accumulated. The probability of a drive not exceeding the mean 90% insertion time by 0.75 seconds is greater than 0.999 for a normal distribution. The measurement of the scram performance of the drives surrounding a drive, which exceeds the expected range of scram performance, will detect local variations and also provide assurance that local scram time limits are not exceeded. Continued monitoring of other drives exceeding the expected range of scram times provides surveillance of possible anomalous performance.

The test schedule provides reasonable assurance of detection of slow drives before system deterioration beyond the limits of Specification 3.3.C. The program was developed on the basis of the statistical approach outlined above and judgement. The occurrence of scram times within the limits, but significantly longer than average, should be viewed as an indication of a systematic problem with control rod drives, especially if the number of drives exhibiting such scram times exceeds eight, which is the allowable number of inoperable rods.

3/4.3.G Control Rod Scram Accumulators

The control rod scram accumulators are part of the control rod drive system and are provided to ensure that the control rods scram under varying reactor conditions. The control rod scram accumulators store sufficient energy to fully insert a control rod at any reactor vessel pressure. The accumulator is a hydraulic cylinder with a free floating piston. The piston separates the water used to scram the control rods from the nitrogen which provides the required energy. The scram accumulators are necessary to scram the control rods within the required insertion times.

Control rods with inoperable accumulators are declared inoperable and Specification 3.3.C then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram than has been analyzed even though control rods with inoperable accumulators may still be inserted with normal drive water pressure. OPERABILITY of the accumulator ensures that there is a means available to insert the control rods even under the most unfavorable depressurization of the reactor.

3/4.3.H Control Rod Drive Coupling

Control rod dropout accidents can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod drop accident is eliminated. Neutron instrumentation response to rod movement may provide verification that a rod is following its drive. Absence of such response to drive movement may indicate an uncoupled condition, or may be due to the lack of proximity of the drive to the instrumentation. However, the overtravel position feature provides a positive check, as only uncoupled drives may reach this position.

3/4.3.I Control Rod Position Indication System (RPIS)

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE. Normal control rod position is displayed by two-digit indication to the operator from position 00 to 48. Each even number is a latching position, whereas each odd number provides information while the rod is in-motion and inputs for rod drift annunciation. The ACTION statement provides for the condition where no positive information is displayed for a large portion or all of the rod's travel. Usually, only one digit of one or two of a rod's positions is unavailable with a faulty RPIS, and the control rod may be located in a known position. However, there are several alternate methods for determining control rod position including the full core display, the four rod display, the rod worth minimizer, and the process computer. Additionally, there are independent "full-in" and "full-out" indicators at the 00 and 48 positions. Another method to determine position would be to move the control rod, by single notch movement, to a position with an OPERABLE position indicator. The original position would then be established and the control rod could be returned to its original position by single notch movement. As long as no control rod drift alarms are received, the position of the control rod would then be known.

3/4.3.J Control Rod Drive Housing Support

The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in Section 4.6.3.5 of the UFSAR. This support is not required if the reactor coolant system is at atmospheric pressure, since there would then be no driving force to rapidly eject a drive housing.

3/4.3.K Scram Discharge Volume Vent and Drain Valves

The scram discharge volume is required to be OPERABLE so that it will be available when needed to accept discharge water from the control rods during a reactor scram and will isolate the reactor coolant system from the containment when required. The operability of the scram discharge volume vent and drain valves assures the proper venting and draining of the volume, so that water accumulation in the volume does not occur. These specifications designate the minimum acceptable level of scram discharge volume vent and drain valve OPERABILITY, provide for the periodic verification that the valves are open, and for the testing of these valves under reactor scram conditions during each refueling outage.

3/4.3.L Rod Worth Minimizer

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. These sequences are developed prior to initial operation of the unit following any refueling outage and the requirement that an operator follow these sequences is supervised by the RWM or a second technically qualified individual. These sequences are developed to limit reactivity worth of control rods and, together with the integral rod velocity limiters and the action of the control rod drive system, limit potential reactivity insertion such that the results of a control rod drop accident will not exceed a maximum fuel energy content of 280 cal/gm. The peak fuel enthalpy of 280 cal/gm is below the energy content at which rapid fuel dispersal and primary system damage have been found to occur based on experimental data. Therefore, the energy deposited during a postulated rod drop accident is significantly less than that required for rapid fuel dispersal.

The analysis of the control rod drop accident was originally presented in Sections 7.9.3, 14.2.1.2, and 14.2.1.4 of the original SAR. Improvements in analytical capability have allowed a more refined analysis of the control rod drop accident which is discussed below.

Every operating cycle the peak fuel rod enthalpy rise is determined by comparing cycle specific parameters with the results of parametric analyses. This peak fuel rod enthalpy is then compared to the analysis limit of 280 cal/gm to demonstrate compliance for that operating cycle. If the cycle specific parameters are outside the range used in the parametric study, an extension of the enthalpy may be required. Some of the cycle specific parameters used in the analysis are: maximum control rod worth, Doppler coefficient, effective delayed neutron fraction and maximum four bundle local peaking factor. The NRC approved methodology listed in Specification 6.6.A.4 provides a detailed description of the methodology used in performing the rod drop analyses.

The rod worth minimizer provides automatic supervision to assure that out-of-sequence control rods will not be withdrawn or inserted, i.e., it limits operator deviations from planned withdrawal sequences (reference UFSAR Section 7.7.2). It serves as a backup to procedural control of control rod worth. In the event that the rod worth minimizer is out-of-service when required, a second licensed operator or other technically qualified individual who is present at the reactor console can manually fulfill the control rod pattern conformance function of the rod worth minimizer. In this case, the normal procedural controls are backed up by independent procedural controls to assure conformance.

3/4.3.M Rod Block Monitor

The rod block monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator, who withdraws control rods according to a written sequence. The specified restrictions with one channel out-of-service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

3/4.3.N Economic Generation Control System

Operation of the facility with the economic generation control system (EGC) (automatic flow control) is limited to the range of 65% to 100% of rated core flow. In this flow range and above 20% of RATED THERMAL POWER, the reactor could safely tolerate a rate of change of load of 8 MWe/sec (reference UFSAR Section 7.7.3.2). Limits within the EGC and the flow control system prevent rates of change greater than approximately 4 MWe/sec. When EGC is in operation, this fact will be indicated on the main control room console.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 137 TO FACILITY OPERATING LICENSE NO. DPR-19, AMENDMENT NO. 131 TO FACILITY OPERATING LICENSE NO. DPR-25, AMENDMENT NO. 158 TO FACILITY OPERATING LICENSE NO. DPR-29, AND AMENDMENT NO. 154 TO FACILITY OPERATING LICENSE NO. DPR-30

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC 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 July 29, 1992, as supplemented by letters dated January 14, 1993, February 16, 1993, and May 9, 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 its Technical Specification Upgrade Program (TSUP). The May 9, 1995, letter provided supplemental information and did not change the staff's initial no significant hazard finding.

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, Revision 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.

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 operation (LCO) and action statements utilizing STS terminology, (3) deleting superseded requirements and modifications to the TS based on the licensee's responses to generic letters (GLs), and (4) relocating specific items to more appropriate TS locations or to licensee controlled documents.

The application dated July 29, 1992, as supplemented January 14, 1993, February 16, 1993, and May 9, 1995, proposed to upgrade only those sections of the TS to be included in TSUP Section 3.3 (Reactivity Controls) 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.

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 licensee's 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 1970s (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 LCOs, 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 1970s. The STS were used as the licensing basis for plants licensed starting in the late 1970s.

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 SR does not change the current design requirements of either plant as described in 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 1980s and early 1990s.

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 licensee's 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 (the "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" ("Final Policy Statement"), 58 FR 39132 (July 22, 1993), in which the Commission indicated that compliance with the Final Policy Statement satisfies Section 182a of the Act. In particular, the Commission indicated that certain items could be relocated from the TS to licensee-controlled documents, consistent with the standard enunciated in *Portland General Electric Co.* (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). In that case, the Atomic Safety and Licensing Appeal Board indicated that "technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety."

Consistent with this approach, 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 or 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 PROPOSED TS SECTION 3/4.3 "REACTIVITY CONTROL"

The following sections provide the staff's evaluation of the TS changes reflected in proposed TS Section 3/4.3, "Reactivity Controls." The current Dresden and Quad Cities TS Section 3.3/4.3 requirements for reactivity limitations have been included within the proposed TS Section 3/4.3. Proposed TS 3/4.3 has been developed in accordance with the guidelines of the STS Section 3/4.1, "Reactivity Controls Systems." The proposed TS were reviewed against the STS guidelines and the current requirements. The deviations between the proposed TS, the current TS, and the STS guidelines are evaluated below.

3.1 TS 3/4.3.A: Shutdown Margin (SDM)

Proposed TS 3/4.3.A, "Shutdown Margin," incorporates the guidelines of STS Section 3/4.1. Plant-specific values for the listed parameters are included to be consistent with the UFSAR for each plant. The proposed TS incorporate the requirements concerning SDM from the Dresden and Quad Cities current TS Section 3.3/4.3.A.

3.1.1 LCO

Proposed TS LCO 3.3.A has been formatted in accordance with the guidelines of the STS Section 3.1.1. A new LCO has been added to the TS in accordance with the STS guidelines. The new LCO requirement specifies demonstration of SDM by analytical methods. The amount of SDM demonstrated analytically is greater than that demonstrated by test (the proposed analytical limit is 0.35% delta k/k). The additional 0.10% delta k/k SDM, based on the STS requirements, is conservative and applicable for Dresden and Quad Cities Stations. In addition, the proposed change to the LCO appropriately places the limiting condition in the LCO instead of the SR as specified in current TS 4.3.A.1. The proposed requirement places appropriate limiting restrictions when analytical methods are used to demonstrate SDM. The staff finds the proposed LCO has incorporated all current TS requirements and enhanced the current requirements. Therefore, the staff finds the proposed LCO requirements for proposed TS Section 3/4.3.A to be acceptable.

3.1.2 Applicability

The current applicability for SDM is implied in current TS 3.3.A.1 and 4.3.A.1 (before making the reactor critical). The applicability in the proposed TS has been expanded in accordance with STS guidelines to include Operational Modes 1, 2, 3, 4, and 5. As a result, operability requirements are added in the proposed TS, for hot shutdown, cold shutdown and refueling which are not explicitly stated in the current TS. This change is an enhancement of current requirements and is therefore acceptable.

3.1.3 Required Actions

The required actions for proposed TS 3.3.A have been formatted in accordance with the STS guidelines. The proposed TS required action statements have incorporated the requirements of the current TS Section 3.3.F for Dresden and 3.3.G for Quad Cities. The current TS required actions specify that the reactor be brought to cold shutdown within 24 hours if SDM cannot be demonstrated. In the proposed TS if the LCO can be met, the action requirements are mode-specific and provide explicit guidance to site operations personnel. In Modes 1 and 2, the proposed TS provide for a 6 hour allowed outage time (AOT) to restore SDM. The current TS does not provide a similar AOT. In the proposed TS, if the SDM requirements cannot be restored within the AOT, the plant is to be in hot shutdown (Mode 3) within the next 12 hours. The proposed TS requirements to bring the plant to hot shutdown conditions within 18 hours following SDM concerns is comparable to current TS requirements to be in cold shutdown within 24 hours. The further restrictions discussed below for Mode 3 (Hot Shutdown) that require all control rods to be fully inserted and all activities adverse to SDM to be suspended compensates for this deviation from the current TS. In terms of safety assurance, hot shutdown is a safer condition with respect to core reactivity than cold shutdown because of the negative moderator temperature coefficient and the greater probability of a cold water insertion event during cold shutdown conditions. For challenges to the plant systems' ability to insert negative reactivity, it is more conservative to have a shorter action period that achieves hot shutdown. Once the unit is in hot shutdown, the reactivity issues have been mitigated. Actions to reach cold shutdown are more important for core decay heat removal or coolant inventory challenges, and the TS that enforce those requirements appropriately mandate actions that continue to cold shutdown. The changes to the required actions in proposed TS Section 3.3 are based on the STS guidelines. The staff finds that the change in the shutdown action requirement when the SDM cannot be met is acceptable. Therefore, the staff finds the required actions in the proposed TS to be acceptable.

3.1.4 Surveillance Requirements

The proposed TS add a new SR in accordance with the STS guidelines. The proposed TS requires the demonstration of SDM within 24 hours after detection of a withdrawn control rod that is immovable as a result of excessive friction or mechanical interference, or is known to be unscrammable. The STS guidelines recommend that the shutdown margin demonstration should be completed within 12 hours of the detection of an immovable control rod. The proposed TS requires completion in 24 hours as a result of the minimum required time to perform the SDM calculations and have them approved in accordance with station procedures. With a single control rod stuck in a withdrawn position, the remaining operable control rods are capable of providing the required scram and shutdown reactivity. Failure to reach cold shutdown is only likely if an additional control rod adjacent to the stuck control rod also fails to insert during a required scram. Even with this postulated additional single failure, sufficient reactivity control remains to reach and maintain hot shutdown conditions. Given these considerations, the staff finds that extending the SDM demonstration to 24 hours in the proposed TS is acceptable and provides a reasonable time to perform the analysis or test. The proposed TS is an enhancement of current TS and is acceptable.

The proposed TS also have added a requirement in accordance with the STS guidelines to verify that the SDM has been demonstrated analytically prior to performing core alterations. This new SR will provide for SDM analytical determination until such time that plant conditions will allow for the SDM demonstration by test. Proposed TS Section 4.3.A.3 adds requirements more restrictive than the current TS requirements. Therefore, the staff finds that the proposed SR are acceptable.

3.1.5 Conclusion

Based on the above evaluation, the staff finds that proposed TS 3/4.3.A, "Shutdown Margin," has been reformatted in accordance with the guidelines of the STS. The staff has reviewed the proposed TS against the STS guidelines and current TS requirements and finds that the deviations from the STS guidelines are acceptable and the relaxations of current TS requirements do not reduce the existing margin of safety. In addition, the proposed TS add requirements which enhance the current TS. Therefore, the staff finds the proposed TS Section 3/4.3.A acceptable.

3.2 TS 3/4.3.B: Reactivity Anomalies

Proposed TS 3/4.3.B, "Reactivity Anomalies," incorporates the guidelines of STS Section 3/4.1.2. Plant-specific values for the listed parameters are included to be consistent with the UFSAR for each plant. The proposed TS incorporate the requirements concerning reactivity anomalies from the Dresden and Quad Cities current TS Section 3.3/4.3.E.

3.2.1 LCO

The current TS requirements concerning reactivity anomalies in Section 3.3/4.3.E of the Dresden and Quad Cities TS have been incorporated into the proposed TS LCO. The proposed TS LCO for Quad Cities deviates from the current TS LCO in that the current TS uses the terminology "critical control rod configuration" and the proposed TS uses the STS terminology "rod density." The terms are equivalent. Therefore the staff finds this deviation to be acceptable. The staff finds that the proposed LCO has incorporated all current TS requirements and has been formatted in accordance with the STS guidelines. Therefore, the staff finds the proposed LCO requirements for proposed TS Section 3/4.3.B to be acceptable.

3.2.2 Applicability

The proposed TS applicability for reactivity anomalies has been formatted in accordance with the STS guidelines and has encompassed the current TS applicability requirements of current TS Section 3.3/4.3.3.E. The current TS requirements for applicability specify power operation only and no other operational modes. The proposed TS requirements clearly define the applicability requirements as Operational Modes 1 and 2 which encompasses power operation. As a result, operability provisions are added to the proposed TS for Startup which is not explicitly stated in the current TS. The staff finds the proposed applicability requirements have encompassed the current TS requirements and enhanced the current TS requirements by making reference to the specific modes of operation. Therefore, the staff finds the proposed applicability statements for proposed TS Section 3/4.3.B to be acceptable.

3.2.3 Required Actions

The required actions proposed for TS 3.3.B have been formatted in accordance with the STS guidelines and incorporate the requirements of TS Section 3.3.E of both the Dresden and Quad Cities TS. The current TS state that if the LCO cannot be met, the reactor must be shut down until the cause of the anomaly has been determined and corrective actions implemented. There are no specific time constraints providing guidance to site operations personnel as to when shutdown shall commence. The proposed TS states that if the LCO cannot be met, an AOT of 12 hours is provided to adequately determine the cause of the anomaly. If justification for continued operation cannot be determined, the reactor is required to be brought to hot shutdown within 12 hours. The proposed requirements provide enhanced guidance to site operations personnel to appropriately disposition potential degraded plant conditions. The proposed action requirements have been shown by industry precedent to provide sufficient guidance to site operations personnel to adequately limit the vulnerability of a plant to potential reactivity concerns. The 12 hour AOT is an adequate amount to time to assess the cause of the anomaly and determine corrective actions without initiating a potentially unnecessary transient. Because the proposed TS provides enhanced guidance to operators without reducing the existing margin of safety, the proposed TS is acceptable.

The current requirements (current TS 3.3.E) to notify the NRC staff within 24 hours in accordance with current TS 6.6 if the LCO cannot be met is beyond the reporting requirements outlined in the STS. This requirement is maintained per 10 CFR 50.72 and 50.73. Therefore, the reporting requirements outlined in current TS Section 3.3.E have not been retained in the proposed TS. The staff has determined that the reporting 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. In addition, the staff finds that sufficient regulatory controls exist under 10 CFR 50.72 and 50.73 to assure continued protection of public health and safety.

The staff finds the required actions in the proposed TS incorporate the requirements of the current TS concerning reactivity anomalies and have been formatted in accordance with the STS guidelines. Therefore, the staff finds the proposed required actions to be acceptable.

3.2.4 Surveillance Requirements

Current TS 4.3.E requirements that discuss performing reactivity anomalies during the (original) startup test program have not been included within proposed TS 4.3.B. These requirements are obsolete since the startup test program has been completed and as such, inappropriate for inclusion in the TS. The staff finds this change acceptable.

Current TS 4.3.E requirements that discuss the comparisons being used as the base data for reactivity monitoring throughout a fuel cycle have not been included in proposed TS 4.3.B. This information is a design detail inappropriate for inclusion as a TS requirement and more appropriately controlled in the UFSARs. The staff has determined that the requirements for base data for reactivity monitoring 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. This change is acceptable.

Current TS 4.3.E requires a comparison of the critical rod configuration to the expected cofiguration. The current TS specify that this comparison be made at power operating conditions (modes 1 and 2). The proposed TS does not include the reference to specific operating conditions since this is implied by the proposed applicability of modes 1 and 2 and it would be redundant to repeat under the surveillance requirements. Therefore, this change does not pose a relaxation from current requirements and is acceptable.

3.2.5 Conclusion

Based on the above evaluation, the staff finds that proposed TS 3/4.3.B, "Reactivity Anomalies," has been reformatted in accordance with the guidelines of the STS. The staff has reviewed the proposed TS against the STS guidelines and current TS requirements and finds that there are no significant deviations from the STS guidelines and the relaxations of current TS requirements do not reduce the existing margin of safety. Therefore, the staff finds the proposed TS Section 3/4.3.B acceptable.

3.3 TS 3/4.3.C: Control Rod Operability

Proposed TS 3/4.3.C, "Control Rod Operability," incorporates the guidelines of STS Section 3/4.1.3.1. Plant-specific values for the listed parameters are included to be consistent with the UFSAR for each plant. The proposed TS incorporate the requirements concerning the operability of control rods from the Dresden and Quad Cities current TS Sections 3.3/4.3.A.

3.3.1 LCO and Applicability

The current TS requirements concerning control rod operability in Section 3.3/4.3.A of the Dresden and Quad Cities TS have been incorporated into the proposed TS LCO and applicability statement. The staff finds the proposed LCO has incorporated all current TS requirements and has been formatted in accordance with the STS guidelines. Therefore, the staff finds the proposed LCO and applicability requirements for proposed TS Section 3/4.3.C to be acceptable.

3.3.2 Required Actions

The current TS require that the plant be brought to cold shutdown within 48 hours if a partially or fully withdrawn control rod drive cannot be moved. The proposed TS deletes this requirement. The applicability for proposed TS 3/4.3.C specifies only Modes 1 and 2 (run and startup). The current TS also specify applicability during reactor power operation (Modes 1 and 2). Therefore, to require a mode change to Mode 4 (cold shutdown) is overly burdensome and unnecessarily introduces a thermal transient to the reactor pressure vessel. If the SR can be satisfied, there is no technical reason why the reactor needs to be shutdown. With a single withdrawn control rod stuck, the remaining operable control rods are capable of providing the required scram and shutdown reactivity. The assumptions utilized in establishing the scram time limits account for a single stuck control rod as well as an additional assumed single failure during a transient. Shutdown margin (SDM) must still be met, which accounts for the loss of negative reactivity due to the stuck control rods. Prompt action is required to confirm no additional stuck control rods exist. Therefore, continued operation can be allowed. The proposed TS eliminates the potential for an unnecessary reactor shutdown and is consistent with the requirements outlined in the Improved STS (NUREG-1433). Based on the above evaluation, this relaxation of current TS requirements is acceptable.

Current TS 3.3.A.2.c contains provisions which specify that control rods that are fully inserted and disarmed may be considered operable. These provisions have been deleted in the proposed TS. Proposed TS section 1.0 provides a general definition of operability which requires that a system, train, or component be capable of performing its specified function. The specific details regarding how this is acheived for each system is not necessary to be included in the TS. These details are controlled administratively by the licensee and their deletion does not represent a relaxation of 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, this change is acceptable.

3.3.3 Surveillance Requirements

Current TS 4.3.A.2 requires that the periodicity of notch exercising be increased to every 24 hours when power operation is continuing with one fully or partially withdrawn rod which cannot be moved and for which control rod drive (CRD) mechanism damage has not been ruled out. This requirement has not been retained in the proposed TS requirements. These specific operability requirements are obsolete and were originally intended as an enhanced method for identifying CRD mechanism damage. CRD mechanism damage and potential failures are detected during control rod disassembly and inspection conducted during regular refueling outages. Therefore, this provision is unnecessary and redundant as a TS requirement. The proposed requirements are consistent with the STS 3.3.C, Actions and the applicable surveillances outlined in STS 4.3.C, which have been shown through industry experience to adequately preclude generic CRD concerns.

Current TS 4.3.A.2 requirements specifying the periodicity of control rod notch exercising to be at least once per 24 hours in the event power operation is continuing with three or more inoperable rods has not been retained in the proposed TS requirements. This additional increased periodicity limitation is obsolete and was originally intended as an enhanced method for identifying CRD mechanism damage. CRD mechanism damage and potential failures are detected during control rod disassembly and inspection conducted during regular refueling outages. The current TS requirement for increased frequency (24 hours) of movement of all partially inserted rods which are movable (whenever greater than three inoperable rods) was intended to shorten the detection time for a hypothetical generic or common-mode failure. This has been shown not to be necessary and is non-conservative for fuel cladding stress cycling whenever three rods are inoperable for any reason, including preventive or elective maintenance that would make rods temporarily inoperable. This is not an uncommon situation where four symmetric rods are inserted for maintenance. The rods may be inoperable until the post-maintenance scram time tests are performed. The current TS would require that all of the other rods then be moved by single notch during the inoperable period. When the next group of four rods was taken out-of-service to continue the work activity, all of the rods would again be single-notch cycled. This repeated control rod movement would add unnecessary power cycles to all of the reactor fuel and increase the potential for mispositioned control rods. Therefore, this provision is proposed for elimination as a TS requirement. The proposed requirements are consistent with the required actions of STS 3.3.C and the applicable surveillances outlined in STS 4.3.C, which have been shown through industry experience to adequately preclude generic CRD concerns. Therefore, this change is acceptable.

Current TS 4.3.A.2 requires increased periodicity of CRD notching if three or more control rods are inoperable or if one control rod is partially withdrawn and for which CRD mechanism damage has not been ruled out. These requirements have not been retained within the proposed TS requirements. Because the proposed TS do not include provisions for increasing the periodicity of CRD notching with three or more inoperable control rods or for which it cannot be shown that the inoperability is a result of CRD housing mechanism failure, this portion of current TS 4.3.2 is no longer applicable. The proposed TS requires increased periodicity of CRD notching with one fully or partially withdrawn control rod. The present reference to a failed CRD collet housing (current TS 4.3.A.2) is outdated and is being deleted from the Dresden and Quad Cities TS. Collet housing cracking and potential failures are detected during control rod disassembly and inspection conducted during regular refueling outages. Therefore, this provision is unnecessary and redundant as a TS requirement and its deletion from proposed TS 3/4.3.C is acceptable.

3.3.4 Conclusion

Based on the above evaluation, the staff finds that proposed TS 3/4.3.C, "Control Rod Operability," has been reformatted in accordance with the guidelines of the STS. The staff has reviewed the proposed TS against the STS guidelines and current TS requirements and finds that there are no significant deviations from the STS guidelines and the relaxations of current TS requirements do not reduce the existing margin of safety. Therefore, the staff finds the proposed TS Section 3/4.3.C acceptable.

3.4 TS 3/4.3.D: Maximum Scram Insertion Times

Proposed TS 3/4.3.D, "Maximum Scram Insertion Times," incorporates the guidelines of STS Section 3/4.1.3.2 and current TS 3/4.3.C. Plant-specific values for the listed parameters are included to be consistent with the UFSAR for each plant.

3.4.1 LCO and Applicability

The current TS requirements concerning maximum scram insertion times in Section 3.3/4.3.C of the Dresden and Quad Cities TS have been incorporated into the proposed TS LCO and applicability statement. The staff finds the proposed LCO has incorporated all current TS requirements and has been formatted in accordance with the STS guidelines. Therefore, the staff finds the proposed LCO and applicability requirements for proposed TS Section 3/4.3.D to be acceptable.

3.4.2 Required Actions

Current Quad Cities TS 3.3.C.4 specifies action in the event that rod(s) do not meet the 90% insertion time required by current TS 3.3.C.2. This action is to insert and disarm the rod. Current Dresden TS do not contain this requirement. Per proposed TS 3.3.D, action 1, the failure to meet the 90% insertion time requirement would make the rod inoperable and the actions for the inoperable rod(s) per proposed TS 3.3.C.2 would then be required. The requirements of proposed TS 3.3.C.2 do not directly require the rod to be inserted but will ensure that the slow rod is separated from any other inoperable, withdrawn controls, and that the rod is moveable with normal CRD drive water pressure. If these requirements are not met, then the rod is required to be inserted and disarmed. This alternative provision for keeping a moderately slow scram time rod withdrawn is justified by the controls specified in proposed TS 3.3.C.2 which limit the number and proximity of such rods. Therefore, the proposed action is acceptable.

Proposed TS 3.3.D, action 2 also requires performance of scram time testing for at least 10% of the control rods at least once per 60 days when operation is continued with three or more control rods with maximum insertion times exceeding 7 seconds. This is a new requirement not in the current TS. The proposed TS is an enhancement of current TS and is therefore acceptable.

3.4.3 Surveillance Requirements

Current TS 4.3.C.1 requires scram time tests on all control rods after a refueling outage and before exceeding 30% power. Proposed TS 4.3.D.1 requires performance of the surveillance prior to exceeding 40% power following core alterations or after a reactor shutdown that is greater than 120 days. The power level is raised in accordance with the STS guidelines and justified because the minimum critical power ratio is non-limiting at the 40% power level. The proposed change in power level from 30% to 40% does not significantly affect existing plant safety margins because adequate margin to the MCPR power distribution limits are experienced at such power levels. In addition, the required controls provided by the Rod Block Monitor (applicable at greater than 30% rated thermal power) provide continued assurance that appropriate controls are in place to prevent inadvertent reactivity excursions that may challenge the MCPR power distribution limits. The higher power level will afford Dresden and Quad Cities additional margin from the banked position withdrawal sequence low power setpoint to perform the scram timing. These changes will continue to ensure testing in situations that can directly affect control rod insertion times; thus, the current plant safety margins are not adversely affected and this change is acceptable.

Proposed TS 4.3.D.2 is a new SR for individual rods following maintenance or modification work that could affect scram times. This is an enhancement of current TS and is acceptable. A footnote is added to the proposed SR to allow operational mode changes prior to performing the required SR for individual control rods that have had maintenance performed. This provides a necessary allowance to bring the reactor to the proper conditions in order to assure scram timing is properly performed yet limits the plant to less than 40% power.

Current TS 4.3.C.2 requires that 50% of the control rods be measured for scram times not more frequently than 16 weeks nor less frequently than 32 weeks. The current requirements are replaced with proposed TS 4.3.D.3 which is based on STS 4.1.3.2.c and requires at least 10% of the control rods, on a rotating basis, to be scram time tested at least once per 120 days of reactor power operation. The scram time testing of proposed TS 4.3.D.3 has been proven successful through use for detecting scram time deterioration at operating boiling water reactors (BWRs) with CRD systems similar in design to that of Dresden and Quad Cities. The population of the control rods subjected to scram timing will be reduced as a result of adopting the STS SR for scram timing, thus reducing unnecessary, excessive wear to the CRDs. The large number of significant control rod moves imposes a large, extended power reduction and movement of many more control rods, with accompanying challenges to cladding (thermal) cycles and control rod positioning. The extent and time of the load drop induces a core xenon transient that further complicates reactor recovery, making the surveillance evolution a significant challenge to the plant and reactivity management while adding minimal data to the extensive performance database. The scram times for all control rods will continue to be measured at the time of each refueling outage per proposed TS 4.3.D.1. Plant experience has shown that control drive insertion times vary little through the operating cycle. The history of drive performance accumulated to date indicates that the 90% insertion times of new and overhauled drives approximate a normal distribution about the mean which tends to become skewed toward longer scram times as operating time is accumulated. The probability of a drive not exceeding the mean 90% insertion time by 0.75 second is greater than 0.999 for a normal distribution. The measurement of the scram performance of the drives surrounding a drive which exceeds the expected range of scram performance will detect local variations and also provide assurance that local scram time limits are not exceeded. Continued monitoring of other drives exceeding the expected range of scram times provides surveillance of possible anomalous performance. The proposed test schedule provides reasonable assurance of detection of slow drives before system deterioration beyond the TS limits. The reduction in the required number of rods tested does not have an adverse effect on the minimum critical power ratio (MCPR) safety limit; thus, the current licensing basis remains unaffected and this change is acceptable.

The provisions in current TS 4.3.C.2 at Quad Cities that require all control rods to be scram tested each year have not been retained within proposed TS 4.3.D. The current Dresden TS do not contain this requirement. As previously discussed, the present Quad Cities requirements are replaced with proposed TS 4.3.D.3 which requires at least 10% of the control rods, on a rotating basis, to be scram time tested at least once per 120 days of reactor power operation. The scram time testing of proposed TS 4.3.D.3 has been proven successful through use for detecting scram time deterioration at operating BWRs with CRD systems similar in design to that of Dresden and Quad Cities. The population of the control rods subjected to scram timing will be reduced as a result of adopting the STS SR for scram timing, thus reducing unnecessary, excessive wear to the CRDs. The staff has determined that the relaxation of current TS requirements maintains the existing safety margin and the proposed SR is therefore acceptable.

Current TS 4.3.C.2 also includes provisions to perform an evaluation after completion of CRD scram tests. This requirement is being deleted since the SR as proposed require, through their performance, evaluations of CRD scram tests. Thus, the evaluations will continue to be performed yet controlled by administrative methods outside of the TS. The current requirement to submit the results of the scram time tests in the annual operating report to the NRC staff has also been deleted. The staff has determined that the requirements for submitting the scram time test results to the NRC 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. Scram time data disposition will continue to be performed; thus, the current licensing basis remains unaffected.

Current TS 4.3.C.3 requires determination of the cycle cumulative mean scram time limit. This has been relocated to proposed TS 3/4.11, "Power Distribution Limits." This is an administrative change which does not reduce current requirements and is acceptable.

The proposed TS also require performance of scram time testing for at least 10% of the control rods at least once per 60 days when operation is continued with three or more control rods with maximum insertion times exceeding 7 seconds. This is a new requirement not in the current TS. The proposed TS is an enhancement of current TS and is therefore acceptable.

3.4.4 Conclusion

Based on the above evaluation, the staff finds that proposed TS 3/4.3.D, "Maximum Scram Insertion Times," has been reformatted in accordance with the STS guidelines. The staff has reviewed the proposed TS against the STS guidelines and current TS requirements and finds that the deviations from the STS guidelines are acceptable and the relaxation of current TS requirements does not reduce the existing margin of safety. Therefore, the staff finds the proposed TS Section 3/4.3.D acceptable.

3.5 TS 3/4.3.E: Average Scram Insertion Times

Proposed TS 3/4.3.E, "Average Scram Insertion Times," incorporates the guidelines of STS Section 3/4.1.3.3 and portions of current TS Section 3/4.3.C. Plant-specific values for the listed parameters are included to be consistent with the UFSAR for each plant.

3.5.1 LCO and Applicability

The current TS requirements concerning average scram insertion times in Section 3.3/4.3.C of the Dresden and Quad Cities TS have been incorporated into the proposed TS LCO and applicability statement. The staff finds the proposed LCO has incorporated all current TS requirements and has been formatted in accordance with the STS guidelines. Therefore, the staff finds the proposed LCO and applicability requirements for proposed TS Section 3/4.3.E to be acceptable.

3.5.2 Required Actions

The current TS require that if the average scram time limit cannot be met, the reactor must be brought to cold shutdown within 24 hours. The proposed TS require that the reactor be brought to hot shutdown with 12 hours. The requirement to bring the plant to hot shutdown rather than cold shutdown provides an equivalent level of safety since the average scram insertion time limits are only required to be met in modes 1 and 2. Therefore, bringing the plant to hot shutdown (mode 3) brings the plant to a condition for which the TS not apply. In addition, the proposed TS requires that the plant be brought to a safe condition sooner (12 hours versus the current 24 hours). Therefore, the proposed TS does not adversely affect existing plant safety margins and is acceptable.

3.5.3 Surveillance Requirements

The proposed TS require that the average scram times shall be demonstrated by scram time testing as required by TS 4.3.D. The deviations between TS 4.3.D and the current requirements are discussed in Section 3.4.3 of this SE and are acceptable.

3.5.4 Conclusion

Based on the above evaluation, the staff finds that proposed TS 3/4.3.E, "Average Scram Insertion Times," has been reformatted in accordance with the STS guidelines. The staff has reviewed the proposed TS against the STS guidelines and current TS requirements and finds that the deviations from the STS guidelines are acceptable and the relaxation of current TS requirements does not reduce the existing margin of safety. Therefore, the staff finds the proposed TS Section 3/4.3.E acceptable.

3.6 TS 3/4.3.F: Group Scram Insertion Times

Proposed TS 3/4.3.F, "Group Scram Insertion," incorporates the guidelines of STS Section 3/4.1.3.4 and portions of current TS 3/4.3.C. Plant-specific values for the listed parameters are included to be consistent with the UFSAR for each plant.

3.6.1 LCO and Applicability

The current TS requirements concerning group scram insertion times in Section 3.3/4.3.C of the Dresden and Quad Cities TS have been incorporated into the proposed TS LCO and applicability statement. The staff finds the proposed LCO has incorporated all current TS requirements and has been formatted in accordance with the STS guidelines. Therefore, the staff finds the proposed LCO and applicability requirements for proposed TS Section 3/4.3.F to be acceptable.

3.6.2 Required Actions

Current TS 3.3.C.3 at Quad Cities has not been retained in proposed TS 3/4.D. The current TS applies to both the slowest rod and average times. The current requirement specifies that if the average scram insertion time for the three fastest control rods of all groups of four control rods cannot be met, the reactor shall not be made supercritical. If operating, the reactor shall be shut down immediately upon determination that the average scram time is deficient. Because the group scram time requirements of proposed TS 3.3.F automatically compensate for the slowest rods (a slow rod is inside four 4-rod groups, and proposed TS 3.3.F requires that the three fastest rods bring the 4-rod average to within the limits of the core analyses), the restriction of current TS 3.3.C.3 makes the provisions of the group average times of current

TS 3.3.C.1 not applicable. This limitation has been removed from STS as it is obsolete and imposes an arbitrary restriction on the control rod system's use. This deviation from current Quad Cities TS is acceptable. The current TS at Dresden do not contain this requirement and therefore, there is no change in the proposed TS.

3.6.3 Surveillance Requirements

The proposed TS require that the average scram times shall be demonstrated by scram time testing as required by proposed TS 4.3.D. The deviations between proposed TS 4.3.D and the current requirements are discussed in Section 3.4.3 of this SE and are acceptable.

3.6.4 Conclusion

Based on the above evaluation, the staff finds that proposed TS 3/4.3.F, "Group Scram Insertion Times," has been reformatted in accordance with the STS guidelines. The staff has reviewed the proposed TS against the STS guidelines and current TS requirements and finds that the deviations from the STS guidelines are acceptable and the relaxation of current TS requirements does not reduce the existing margin of safety. Therefore, the staff finds the proposed TS Section 3/4.3.F acceptable.

3.7 TS 3/4.3.G: Control Rod Scram Accumulators

Proposed TS 3/4.3.G, "Control Rod Scram Accumulators," incorporates the guidelines of STS Section 3/4.1.3.5 and current TS 3/4.3.D. Plant-specific values for the listed parameters are included to be consistent with the UFSAR for each plant.

3.7.1 LCO

Current TS 3.3.D includes information that defines the operability of the system. These details have been deleted in the proposed TS. Proposed TS section 1.0 provides a general definition of operability which requires that a system, train, or component be capable of performing its specified function. The specific details regarding how this is acheived for each system is not necessary to be included in the TS. These details are controlled administratively by the licensee and their deletion does not represent a relaxation of 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, this change is acceptable.

Current TS 3.3.D allows one rod accumulator to be inoperable provided that no other control rod in the nine-rod square array around this rod is inoperable for a variety of reasons. The proposed TS have deleted this provision and require all rod accumulators to be operable. If a rod accumulator is inoperable, its associated control rod must be declared inoperable. The proposed TS for inoperable control rods (proposed TS 3.3.C) requires that the inoperable control rod, if withdrawn, be seperated from all other inoperable withdrawn control rods by at least two control cells in all directions. Therefore, the current use of the nine-rod square array has been replaced by the designation of two control cells. The current and proposed TS provide an equivalent level of safety for assuring that inoperable control rods are seperated by an adequate distance to maintain reliable reactivity control. Therefore, the proposed TS is acceptable.

3.7.2 Required Actions

In the event that the LCO requirements cannot be met, current TS 3.3.F for Dresden and 3.3.G for Quad Cities specify that the reactor be brought to cold shutdown within 24 hours. The proposed TS action requirements specify that the reactor shall be brought to hot shutdown within 12 hours. The requirement to bring the plant to hot shutdown rather than cold shutdown provides an equivalent level of safety since the control rod scram accumulators are only required to be operable in modes 1 and 2 and under certain circumstances in mode 5. Therefore, bringing the plant to hot shutdown (mode 3) brings the plant to a condition for which the TS not apply. In addition, the proposed TS requires that the plant be brought to a safe condition sooner (12 hours versus the current 24 hours). Therefore, the proposed TS does not adversely affect existing plant safety margins and is acceptable.

The current TS contain an exception that allows the rod block associated with an inoperable accumulator to be bypassed if the control rod is inserted full-in and its directional control valves are electrically disarmed. This exception is deleted because the proposed specifications require the inoperable accumulator to be made operable within 8 hours or the control rod to be fully inserted and declared inoperable. When the control rod is fully inserted and inoperable, the allowance for a control rod block bypass can be removed because the control rod is secured and this information does not need to be included in the LCO. Therefore, this change is acceptable.

Proposed TS 3.3.G, action 1.c(1) is an additional requirement not in current TS. The proposed TS specifies that if the control rod associated with an inoperable accumulator is withdrawn, one CRD pump must be immediately verified to be operable. With no CRD pump operating, an immediate scram of the reactor is required. The proposed TS are more restrictive with more limiting action provisions than current TS. In addition, specific time constraints are included in TS 3.3.G, Actions that provide explicit guidance to site operations personnel and have been shown by industry experience to adequately limit plant vulnerabilities to accumulator concerns. This change is an enhancement of current TS and is therefore acceptable.

Proposed TS 3.3.G, action 2 provides additional requirements for inoperable scram accumulators in Operational Mode 5. The proposed requirements are consistent with STS 3.1.3.5 Actions and conservatively provide additional requirements not part of the current TS for Dresden and Quad Cities. The proposed requirements have been determined to be applicable to the Dresden and

Quad Cities CRD scram accumulator design and provide additional assurances that conservative and appropriate actions are enacted during refueling operations with potentially degraded CRD scram accumulators. These changes are an enhancement of current requirements and are acceptable.

3.7.3 Surveillance Requirements

The current TS requires a shiftly (current TS shiftly is once per 8 hours) check of the status of the pressure and level alarms for each accumulator. Proposed TS 4.3.G specifies an operability check once per 7 days by verifying indicated pressure is greater than or equal to 800 psig unless the control rod is fully inserted, disarmed, or scrammed. The specific operability limitations ensure adequate guidance is provided to site operating personnel. Although the current TS requirement has been relaxed from shiftly to every 7 days, the proposed periodicity is consistent with STS guidelines and has been shown based on use at other plants to provide adequate indication of scram accumulator operability. Therefore, this change is acceptable.

STS 4.1.3.5.b.1 regarding the control rod accumulator leak detectors, pressure, and alarm surveillances have not been adopted as it is not part of the current licensing basis. The scram accumulator leak detectors, pressure detectors, and associated alarms do not necessarily relate directly to accumulator operability. This deviation from STS is consistent with the Improved STS (NUREG-1433) which does not specify indication-only or test equipment to be operable to support operability of a system or component. These activities are addressed by plant operational procedures and policies. In addition, STS TS 4.1.3.5.b.2 has not been adopted as it is not part of the current licensing basis. There is no accident or transient analytical assumption that the control rod accumulator check valves maintain accumulator pressure for a specified time period should no CRD pump be operating. With no operating CRD pump, the reactor must be scrammed within a short time if a pump is not restored to operation. Therefore, STS SR 4.1.3.5.b.2 is not required and any testing for this system is to be controlled in administrative procedures. These deviations are acceptable.

3.7.4 Conclusion

Based on the above evaluation, the staff finds that proposed TS 3/4.3.G, "Control Rod Scram Accumulators," has been reformatted adopting the STS guidelines. The staff has reviewed the proposed TS against the STS guidelines and current TS requirements and finds that the deviations from the STS guidelines are acceptable and the relaxation of current TS requirements does not reduce the existing margin of safety. Therefore, the staff finds the proposed TS Section 3/4.3.G acceptable.

3.8 TS 3/4.3.H: Control Rod Drive Coupling

Proposed TS 3/4.3.H, "Control Rod Drive Coupling," incorporates the guidelines of STS Section 3/4.1.3.6 and current TS Section 3/4.3.B. Plant- specific values for the listed parameters are included to be consistent with the UFSAR for each plant.

3.8.1 LCO and Applicability

The current TS requirements concerning CRD coupling in Section 3.3/4.3.B of the Dresden and Quad Cities TS have been incorporated into the proposed TS LCO and applicability statement. The staff finds the proposed LCO has incorporated all current TS requirements and has been formatted in accordance with the STS guidelines. Therefore, the staff finds the proposed LCO and applicability requirements for proposed TS Section 3/4.3.H to be acceptable.

3.8.2 Required Actions

Current TS 3.3.B.1.a at Dresden specifies that below 20% power, for a rod not coupled to its drive, the control rod shall be declared inoperable and the rod fully inserted with the directional control valves disarmed until recoupling can be attempted above 20% power. Current TS 3.3.B.1.b at Dresden specifies that above 20% power, recoupling is required to be attempted in accordance with established procedures or the rod remains inoperable, fully inserted, and disarmed. Current TS 3.3.B.1.b at Dresden is encompassed within proposed TS 3.3.H, Action 1.b, which requires recoupling attempts to be attempted if permitted by the RWM. This is consistent with the current Dresden requirement of above 20% power. Therefore, the proposed TS is consistent with current Dresden TS and is acceptable.

Current Quad Cities TS 3.3.B.1 does not provide for actions to recouple the control rod. Therefore, the proposed action is an enhancement to the current TS and is acceptable.

Additional actions have been added in Mode 5 for CRD coupling based on the guidance provided in STS 3.1.3.6, Action b. The proposed requirements conservatively place additional restrictions to ensure that appropriate controls are in place during refueling operations that limit the plant to vulnerabilities associated with CRD coupling. The proposed changes have been shown, based on industry precedence, to provide an adequate level of protection from CRD coupling concerns during Mode 5. This change is an enhancement to current TS and is acceptable.

Quad Cities current TS contain a provision which allows two CRDs to be removed as long as shutdown margin (SDM) can be maintained. Allowances for CRD maintenance have been relocated to TSUP Sections 3/4.10.I and 3/4.10.J which continue to ensure that SDM requirements are maintained. This change is administrative and does not reduce existing requirements and is therefore acceptable.

The current Quad Cities TS also contain an exception during periods when the reactor is vented. This has not been retained within the proposed TS. The deletion of this allowance is conservative as it updates Quad Cities to current industry standards that removes a nonconservative allowance and ensures appropriate controls on the CRD system are maintained. This change is an enhancement of current TS and is acceptable.

3.8.3 Surveillance Requirements

Current TS 4.3.B.1.a.i, ii, and iii for Dresden are encompassed within proposed TSUP 4.3.H.1. There are no comparable requirements in current Quad Cities TS. Therefore, the proposed TSUP requirements maintain the same level of protection at Dresden and provide additional limitations at Quad Cities that have been shown by Dresden's experience to provide an adequate level of protection against CRD coupling concerns. This is an enhancement to current Quad Cities TS and is acceptable.

3.8.4 Conclusion

Based on the above evaluation, the staff finds that proposed TS 3/4.3.H, "Control Rod Drive Coupling," has been reformatted adopting the STS guidelines. The staff has reviewed the proposed TS against the STS guidelines and current TS requirements and finds that the deviations from the STS guidelines are acceptable and there are no relaxations of current TS requirements. Therefore, the staff finds the proposed TS Section 3/4.3.H acceptable.

3.9 TS 3/4.3.I: Control Rod Position Indication

Proposed TS 3/4.3.I, "Control Rod Position Indication," incorporates the guidelines of STS Section 3/4.1.3.7. Plant-specific values for the listed parameters are included to be consistent with the UFSAR for each plant. The proposed TS incorporate the requirements concerning the control rod position indication from the Quad Cities current TS Sections 3.3/4.3.A. The current Dresden TS do not contain requirements for control rod position indication.

3.9.1 LCO and Applicability

The current TS requirements concerning control rod position indication in Section 3.3/4.3.A of the Quad Cities TS have been incorporated into the proposed TS LCO. The staff finds the proposed LCO has incorporated all current TS requirements and has been formatted in accordance with the STS guidelines. Therefore, the staff finds the proposed LCO requirements for proposed TS Section 3/4.3.I to be acceptable.

The applicability specified in proposed TS Section 3.3.I incorporates the applicability of STS Section 3.1.3.7. Although not explicitly discussed in Quad Cities current TS 3/4.3.A.3, the current applicability for control rod position is in the startup or run modes. The applicability is expanded in accordance with STS requirements to include withdrawn control rods in Operational Mode 5. As a result, operability requirements are added for refuel which are not in the current TS. This change is an enhancement to current requirements and is therefore acceptable.

3.9.2 Required Actions

In the event that the LCO cannot be met, current TS 3.3.G for Quad Cities specifies that the reactor be brought to cold shutdown within 24 hours. Proposed TS 3.3.I requires the plant to be in hot shutdown within 12 hours. The current and proposed TS requirements ensure that the reactor is brought to a safe condition for which the operability of the degraded equipment is not required. Achieving hot shutdown (Mode 3) in 12 hours ensures that action will be taken sooner and is therefore more restrictive than achieving cold shutdown (Mode 4) in 24 hours. There is no need to bring the plant to cold shutdown since the proposed applicability only requires the position indication system to be operable in modes 1 and 2 and mode 5 for certain rods. Because operability is not required in hot shutdown (mode 3), this is a safe condition for the plant to be in with an inoperable position indicator. In addition, the proposed changes eliminate the unnecessary thermal transient on the reactor pressure vessel. The proposed requirements are consistent with STS guidelines and have been shown based on industry experience to adequately limit plant vulnerabilities to control rod position indication concerns. Therefore, the proposed TS is acceptable.

3.9.3 Surveillance Requirements

Current provisions within the Quad Cities TS (4.3.A.3.a) have been changed from performing shiftly (every 8 hours) checks of the rod position indication system (RPIS) to once per 24 hours. This requirement is not included in current Dresden TS. The intent of the current requirement is to ensure that position indication is determined on a frequency that will ensure operability of the system. The periodicity included in the STS and proposed TS is consistent with industry precedent and has been shown to be acceptable to adequately ensure RPIS operability. In addition, the provisions specified in current Quad Cities TS 4.3.A.3.b requiring all control rods that are fully inserted and scrammed to be given an insert signal once per shift has been deleted in TSUP Section 4.3.I. These requirements are not included in the STS or the Improved STS. These requirements are obsolete and add additional exercise of the CRD system which is unnecessary. In addition, the guidance in GL 93-05, "Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operations," eliminates requirements to perform unnecessary surveillances that make systems less reliable. Therefore, the deletion of current Quad Cities requirement 4.3.A.3.b is acceptable.

3.9.4 Conclusion

Based on the above evaluation, the staff finds that proposed TS 3/4.3.I, "Control Rod Position Indication," has been reformatted in accordance with the guidelines of the STS, and the deviations from the current TS requirements do not reduce the margin of safety for Quad Cities. The proposed TS is an enhancement of current Dresden TS which currently has no requirements. Therefore, the staff finds proposed TS Section 3/4.3.I to be acceptable.

3.10 TS 3/4.3.J: Control Rod Drive Housing Support

Proposed TS 3/4.3.J, "Control Rod Drive Housing Support," incorporates the guidelines of STS Section 3/4.1.3.8. Plant-specific values for the listed parameters are included to be consistent with the UFSAR for each plant. The proposed TS incorporate the requirements concerning the CRD housing support from the Dresden and Quad Cities current TS Section 3.3/4.3.B.

3.10.1 LCO

The proposed LCO has incorporated all the current TS requirements from current TS Section 3.3.B.2 at Dresden and 3.3.C.3 for Quad Cities concerning CRD housing Supports. The staff finds the proposed LCO has incorporated all the current TS requirements and has been formatted in accordance with STS guidelines. Therefore, the staff finds the proposed LCO requirements for proposed TS Section 3/4.3.J to be acceptable.

3.10.2 Applicability

The current applicability for the CRD housing support is during reactor power operation and when the reactor coolant system is pressurized above atmospheric pressure with fuel in the vessel unless all control rods are fully inserted and SDM requirements are maintained. The applicability is equivalent to STS requirements (Modes 1, 2, and 3) as encompassed within the proposed TS. The proposed TS conservatively do not include the current applicability exception that allows for inoperable CRD housing supports. This change is an enhancement of STS requirements. The staff finds the proposed applicability requirements have encompassed the current TS requirements and enhanced the current TS requirements by making reference to the specific modes of operation. Therefore, the staff finds the proposed applicability statements for proposed TS Section 3/4.3.J to be acceptable.

3.10.3 Required Actions

The current TS requirements specify that the reactor be brought to cold shutdown within 24 hours if the LCO cannot be met. The proposed TS actions specify that, in the event the proposed TS LCO requirements cannot be met, the plant must be in at least hot shutdown within 12 hours and cold shutdown within the following 24 hours. The proposed TS requirements are in accordance with the STS guidelines. The proposed TS requires that a plant shutdown be initiated sooner than the current TS requirements, upon discovery of the degraded condition, to satisfy the proposed TS action statement. Therefore, although the proposed TS relax the time that is allowed to bring the reactor to cold shutdown from 24 to 36 hours, the proposed TS ensure that the plant is brought out of the operating modes sooner and provides sufficient time for a controlled shutdown to mode 4. Therefore, the staff finds the proposed required actions to be acceptable.

3.10.4 Surveillance Requirements

The proposed TS deviates from current requirements by not retaining the specific requirement to record the results of the CRD housing support inspection. The recording of TS SR is encompassed within the administrative controls of TS Section 6.0 and, as such, is unnecessary for repetition in proposed TS Section 4.3.J. The staff finds the elimination of the reporting requirement to be acceptable. The proposed TS SR have been formatted in accordance with the STS guidelines and have incorporated the existing current TS SR with the exception of the redundant reporting requirement. Therefore, the staff finds the proposed TS SR to be acceptable.

3.10.5 Conclusion

Based on the above evaluation, the staff finds that proposed TS 3/4.3.J, "Control Rod Drive Housing Support," has been reformatted adopting the STS guidelines. The staff has reviewed the proposed TS against the STS guidelines and current TS requirements and finds that there are no significant deviations from the STS guidelines and the relaxations of current TS requirements do not reduce the existing margin of safety. Therefore, the staff finds the proposed TS Section 3/4.3.J acceptable.

3.11 TS 3/4.3.K: SDV [Scram Discharge Volume] Vent and Drain Valves

Proposed TS 3/4.3.K, "SDV Vent and Drain Valves," has been formatted in accordance with the STS guidelines of Section 3/4.1.3.1. Plant-specific values for the listed parameters are included to be consistent with the UFSAR for each plant. The proposed TS incorporate the requirements concerning SDV vent and drain valves from the Dresden and Quad Cities current TS Sections 3.3/4.3.A.3 for Dresden and 3.3/4.3.B.6 for Quad Cities. The current TS contain no LCO, applicability or required action statements for the SDV vent and drain valves, only SR.

3.11.1 LCO

Proposed TS 3/4.3.K, "Scram Discharge Volume (SDV)," has been formatted in accordance with the STS guidelines contained in STS Section 3/4.1.3.1. These LCO requirements are an enhancement of the current TS which provide no requirements, and are therefore acceptable.

3.11.2 Applicability

The applicability specified in proposed TS Section 3.3.K incorporates the applicability of STS Section 3.1.3.1. Although not explicitly discussed in Dresden current TS 3/4.3.A.3 and Quad Cities current TS 4.3.B.6, the present applicability for SDV is in the startup, run modes and the refuel mode. The proposed TS requirements clarify existing limitations that are consistent to the current design basis, and the proposed changes do not reduce existing plant safety margins.

3.11.3 Required Actions

Proposed TS 3.3.K requires the plant to be in hot shutdown within 12 hours if the LCO cannot be met. The proposed TS requirements ensure the reactor is brought to a safe condition for which the operability of the degraded equipment is not required. This requirement is an enhancement of current TS which provides no requirements and is therefore acceptable.

3.11.4 Surveillance Requirements

The proposed SR for TS Section 3/4.3.K incorporate the requirements of current TS 3/4.3.A.3 for Dresden and 3.3/4.3.B.6 for Quad Cities. The proposed SR are consistent with the current requirements and are therefore acceptable.

The proposed TS do not adopt STS SR 4.1.3.1.4.c and STS footnote *. In addition, the requirement to perform STS SR 4.1.3.1.4.a at 50% rod density was not adopted. The operability of the SDV valves can be satisfactorily demonstrated during a scram from shutdown conditions. The lower coolant temperatures expected during testing at shutdown conditions will also have a negligible impact on the performance of the surveillance. Therefore, since this surveillance can be satisfactorily performed at shutdown conditions, the 50% control rod density requirement is not required. This deviation from STS is acceptable.

The STS requirements for the functional testing of the SDV level sensors (TS 4.1.3.1.4.b) have been relocated to proposed TS Sections 3/4.1 and 3/4.2. This change is administrative and is therefore acceptable.

3.11.5 Conclusion

Based on the above evaluation, the staff finds that proposed TS 3/4.3.K, "SDV Vent and Drain Valves," has been reformatted adopting the STS guidelines. The staff has reviewed the proposed TS against the STS guidelines and current TS requirements and finds that the deviations from the STS guidelines are acceptable and there are no relaxations of current requirements. Therefore, the staff finds the proposed TS Section 3/4.3.K acceptable.

3.12 TS 3/4.3.L: Rod Worth Minimizer (RWM)

Proposed TS 3/4.3.L, "Rod Worth Minimizer," has been formatted in accordance with the guidelines of STS Section 3/4.1.4.1. Plant-specific values for the listed parameters are included to be consistent with the UFSAR for each plant. The proposed TS incorporate the requirements concerning the RWM from the Dresden and Quad Cities current TS Sections 3.3/4.3.B.

3.12.1 LCO

Proposed TS LCO 3.3.L has been formatted in accordance with the guidelines of the STS. The proposed LCO has incorporated all current TS requirements.

Therefore, the staff finds the proposed LCO requirements for proposed TS Section 3/4.3.L to be acceptable.

3.12.2 Applicability

The proposed Quad Cities TS specifies applicability as Modes 1 and 2 when thermal power is less than or equal to 10% of rated thermal power. The current low power setpoint for Quad Cities is 20%. The proposed low power setpoint for Quad Cities is based on "NRC Safety Evaluation Report Approving Amendment 17 to NEDE-24011-P, (GESTR)," dated December 27, 1987. The safety evaluation was issued in response to a topical report submitted by the BWR Owners Group. The topical report proposed lowering the low power setpoint from 20% to 10%. The change to a low power setpoint of 10% reactor power is an important reactivity management improvement because use of a 20% low power setpoint significantly increases axial peaking in the reactor core due to the large number of control rod tips banked together at the specified positions in the core. The NRC found the report acceptable for referencing in license applications. Therefore, the staff finds that lowering the Quad Cities low power setpoint from 20% to 10% is acceptable. Siemens Nuclear currently performs the neutronic analysis for Dresden Station and thus, the topical report is not applicable to Dresden. Therefore, the proposed setpoint for Dresden is consistent with the current setpoint of 20%.

The current TS allow the RWM to be bypassed for low power physics testing if a nuclear engineer is present and verifies the rod movements of the test procedure. The specific requirements for a nuclear engineer to be present have been relocated to operating procedures. The staff has determined that the requirements for a nuclear engineer to be present 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 removal of this requirement from TS is acceptable.

3.12.3 Required Actions

Those portions of current TS 3.3.B.3.b that prescribe that at least 12 control rods must be fully withdrawn before a second operator or qualified technical person can be used as a substitute for the RWM have not been retained in the proposed TS. Dresden and Quad Cities have installed highly reliable RWM computer hardware which has minimized the frequency of reliance on a second verifier as the only backup check to the manual rod movements. In addition, the new RWM is flexible enough to allow sequence changes as needed which prevents the need to bypass it for evolutions that do not match the loaded sequence. Based upon industry precedent specified in the STS, the proposed action has been demonstrated to provide adequate assurance that control rods will be withdrawn in accordance with prescribed patterns (with the necessity of requiring 12 control rods to be fully withdrawn before a substitute can be used for the RWM as unnecessary). The staff finds eliminating this requirement from the existing TS to be acceptable.

Current TS 3.3.B.3.a describes how control rod sequences are to be established to limit maximum reactivity addition so that the rod drop accident design limit of 280 cal/gm is not exceeded. This type of information is inappropriate for inclusion within the TS but is more appropriately located in the bases of the TS. The proposed LCO assures that the RWM is operable and the RWM limits rod movement such that if a rod drop event occurred, the maximum reactivity addition would not exceed 280 cal/gm. Therefore, this information has been relocated to the bases section of the proposed TS. The staff finds this acceptable.

3.12.4 Surveillance Requirements

The proposed SR for proposed TS Section 3/4.3.L are adopted from the STS guidelines. Current TS 4.3.B.3.a.i for Dresden and current TS 4.3.B.3 for Quad Cities is encompassed within proposed TS 4.3.L.1. The proposed requirements continue to assure that the control rod sequences are loaded into the RWM computer and verified to be correct.

The current TS require that the RWM computer on line diagnostic test be performed. The proposed TS do not specify a diagnostic test for verifying operability of the RWM computer. Specifying the diagnostic test of the RWM is redundant to the definition of operability required by the LCO for TS 3.3.L. As such, the specific reference to an RWM computer test is inappropriate for inclusion within TSUP and has been relocated to procedural controls. In addition, the computer verifications required by TS 4.3.L provide continued assurance that the RWM computer is properly controlling out-of-sequence rod maneuvers. This information is a design detail inappropriate for inclusion as a TS requirement and more appropriately controlled in the station procedures. The staff has determined that the requirements for diagnostic testing of the RWM 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.

The proposed TS contain additional restrictions to verify that the RWM is operable when reducing power below 20% (Dresden) and 10% (Quad Cities). There are no current explicit current requirements to govern this maneuver. The proposed requirements are consistent with STS and have shown through industry experience to provide an adequate level of protection for a potential rod drop accident. This change is an enhancement to the current TS and is acceptable.

3.12.5 Conclusion

Based on the above evaluation, the staff finds that proposed TS 3/4.3.L, "Rod Worth Minimizer," has been reformatted in accordance with the guidelines of the STS and the deviations from the current TS requirements do not reduce the margin of safety for Dresden or Quad Cities. Therefore, the staff finds proposed TS Section 3/4.3.L to be acceptable.

3.13 TS 3/4.3.M: Rod Block Monitor (RBM)

Proposed TS 3/4.3.M, "Rod Block Monitor," incorporates the guidelines of STS Section 3/4.1.4.3. Plant-specific values for the listed parameters are included to be consistent with the UFSAR for each plant. The proposed TS incorporate the requirements concerning the RBM from the Dresden and Quad Cities current TS Section 3.3/4.3.B.

3.13.1 LCO

Proposed TS LCO 3.3.M has been formatted in accordance with the guidelines of the STS. The proposed LCO has incorporated all current TS requirements. Therefore, the staff finds the proposed LCO requirements for proposed TS Section 3/4.3.M to be acceptable.

3.13.2 Applicability

The current TS required the rod block to be operable only during limiting control rod patterns. The proposed applicability is operational mode 1 when thermal power is greater than or equal to 30% of rated thermal power. The current TS definition of a limiting control rod pattern is vague and provides limited restrictions on rod block monitor unavailability. The present definition requires the plant to be operating with a control rod pattern that is beyond reasonable design basis assumptions. Therefore, the probability of operating with a limiting control rod pattern, as is currently defined, is extremely unlikely. The proposed applicability enhances current requirements by requiring the operability of the rod bock monitor for an extended range of operating regions. The proposed applicability is an enhancement of current requirements and is therefore acceptable.

3.13.3 Required Actions

The current TS definition of a limiting control rod pattern is vague and provides limited restrictions on RBM unavailability. The current definition assumes the plant to be operating with a control rod pattern that is beyond reasonable design-basis assumptions (requiring multiple control rod withdrawal errors as a result of multiple human errors). Therefore, the probability of operating with a limiting control rod pattern, as is currently defined, is extremely unlikely. The proposed TS definition of limiting control rod pattern is consistent to STS requirements which specifies a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR [average planar linear heat generation rate], LHGR [linear heat generation rate], or MCPR. The proposed TS actions are consistent to STS 3.1.4.3, Actions. The proposed action requirements will continue to ensure that the RBM prevents fuel damage in the event of an erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided, one of which may be bypassed for maintenance or testing. Tripping of one of these channels will block erroneous rod withdrawal in sufficient time to prevent fuel damage. This system backs up the operator, who is required to withdraw control rods according to a written sequence. The specified restrictions with one channel out-of-service conservatively assure that fuel damage will not occur due to rod withdrawal errors. Therefore, the proposed TS actions are acceptable.

Current RBM action requirements do not include specific time limitations for which actions are required as compared to those in the proposed TS. The proposed action requirements provide more specific time limitations as compared to the current TS and provide site operations personnel with enhanced guidance for dispositioning potentially degraded RBM conditions.

3.13.4 Conclusion

Based on the above evaluation, the staff finds that proposed TS 3/4.3.M, "Rod Block Monitor," provides an equivalent level of protection when compared to current requirements. Any deviations from the current TS requirements do not reduce the margin of safety for Dresden or Quad Cities. Therefore, the staff finds proposed TS Section 3/4.3.M to be acceptable.

3.14 TS 3/4.3.N: Economic Generation Control (EGC) System

Proposed TS 3/4.3.N, "Economic Generation Control (EGC) System," incorporates the requirements of current TS Section 3/4.3.G for Dresden and 3/4.3.F for Quad Cities. Plant-specific values for the listed parameters are included to be consistent with the UFSAR for each plant. There are no STS guidelines for EGC.

3.14.1 LCO and Applicability

The current TS requirements concerning the EGC system in Section 3/4.3.G of the Dresden TS and Section 3.3/3.4.F of the Quad Cities TS have been incorporated into the proposed TS LCO and applicability statement. The staff finds the proposed LCO has incorporated all current TS requirements. Therefore, the staff finds the proposed LCO and applicability requirements for proposed TS Section 3/4.3.N to be acceptable.

3.14.2 Surveillance Requirements

Current TS 4.3.G (Dresden) and 4.3.F (Quad Cities) require that the EGC system operating parameters be reviewed for acceptability prior to entering EGC and once per shift. Current practices at Dresden and Quad Cities define a shift to be 8 hours. Proposed TS 4.3.N specifies that the EGC system be demonstrated operable prior to entering EGC and at least once per 12 hours. Although the proposed TS requirement deviates from current TS, it provides more explicit guidance to site operations personnel. For example, the current TS for EGC do not provide any guidance regarding actions and the current SR provides a vague requirement that EGC will be reviewed for acceptability. The proposed SR specifies that the EGC system shall be demonstrated operable and operability requirements are defined and applied consistently to plant systems. Therefore, the proposed SR requires additional actions be taken and provides the operators with more useful information regarding the operability of the system. Performing the proposed surveillance twice per day as opposed to performing the current surveillance three times per day, provides an equivalent or better determination of system condition that will allow operators to adequately address potential degraded conditions. Therefore, the proposed TS is acceptable.

3.14.3 Conclusion

Based on the above evaluation, the staff finds that proposed TS 3/4.3.N, "Economic Generation Control (EGC) System," provides an equivalent level of protection when compared to current requirements. Any deviations from the current TS requirements do not reduce the margin of safety for Dresden or Quad Cities. Therefore, the staff finds proposed TS Section 3/4.3.N to be acceptable.

3.15 Technical Specification Bases

The staff has reviewed the proposed bases for TS Section 3/4.3. The proposed bases have been prepared using the guidelines of the STS. The staff finds these proposed bases acceptable.

4.0 SUMMARY

The proposed TS for Section 3/4.3, "Reactivity Control," will be clearer and easier to use as a result of the adaptation 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 these amendments 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

(58 FR 34071). 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.

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