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. 143
License No. DPR-19

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

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated September 17, 1993, as supplemented by letter dated July 20, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

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(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 143 , 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

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: November 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. 137
License No. DPR-25

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

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated September 17, 1993, as supplemented by letter dated July 20, 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. 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 June 30, 1996.

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 27, 1995

ATTACHMENT TO LICENSE AMENDMENT NOS. 143 AND 137

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
3/4.7-1	3/4.7-1	3/4.7-1
3/4.7-2	3/4.7-2	3/4.7-2
3/4.7-3	3/4.7-3	3/4.7-3
3/4.7-4	3/4.7-4	3/4.7-4
3/4.7-5	3/4.7-5	3/4.7-5
3/4.7-6	3/4.7-6	3/4.7-6
3/4.7-7	3/4.7-7	3/4.7-7
3/4.7-8	3/4.7-8	3/4.7-8
3/4.7-9	3/4.7-9	3/4.7-9
3/4.7-9a	3/4.7-9a	3/4.7-10
3/4.7-10	3/4.7-10	3/4.7-11
3/4.7-11	3/4.7-11	3/4.7-12
3/4.7-12	3/4.7-12	3/4.7-13
3/4.7-13	3/4.7-13	3/4.7-14
3/4.7-14	3/4.7-14	3/4.7-15
3/4.7-15	3/4.7-15	3/4.7-16
3/4.7-16	3/4.7-16	3/4.7-17
3/4.7-17	3/4.7-17	3/4.7-18
3/4.7-18	3/4.7-18	3/4.7-19
3/4.7-19	3/4.7-19	3/4.7-20
3/4.7-19a	3/4.7-19a	3/4.7-21
3/4.7-19b	3/4.7-19b	3/4.7-22
3/4.7-20	3/4.7-20	3/4.7-23
3/4.7-21	3/4.7-21	3/4.7-24
3/4.7-22	3/4.7-22	3/4.7-25
3/4.7-23	3/4.7-23	B 3/4.7-1
3/4.7-24	3/4.7-24	B 3/4.7-2
3/4.7-25	3/4.7-25	B 3/4.7-3
3/4.7-26	3/4.7-26	B 3/4.7-4
3/4.7-27	3/4.7-27	B 3/4.7-5
3/4.7-28	3/4.7-28	B 3/4.7-6
3/4.7-29	3/4.7-29	B 3/4.7-7
3/4.7-30	3/4.7-30	B 3/4.7-8
3/4.7-31	3/4.7-31	---
3/4.7-32	3/4.7-32	---
B 3/4.7-33	B 3/4.7-33	---
B 3/4.7-33a	B 3/4.7-33a	---
B 3/4.7-34	B 3/4.7-34	---
B 3/4.7-35	B 3/4.7-35	---
B 3/4.7-36	B 3/4.7-36	---
B 3/4.7-37	B 3/4.7-37	---

UNIT 2 REMOVE	UNIT 3 REMOVE	INSERT
B 3/4.7-38	B 3/4.7-38	---
B 3/4.7-39	B 3/4.7-39	---
B 3/4.7-40	B 3/4.7-40	---
B 3/4.7-41	B 3/4.7-41	---
B 3/4.7-42	B 3/4.7-42	---
B 3/4.7-43	B 3/4.7-43	---
B 3/4.7-43a	B 3/4.7-43a	---
B 3/4.7-44	B 3/4.7-44	---
B 3/4.7-45	B 3/4.7-45	---
B 3/4.7-46	B 3/4.7-46	---
B 3/4.7-47	B 3/4.7-47	---

3.7 - LIMITING CONDITIONS FOR OPERATION

A. PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY:

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

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

A. PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

1. After each closing of each penetration subject to Type B testing, except the primary containment air locks, if opened following Type A or B test, by leak rate testing the seals with gas at $\geq P_a$ (48 psig), and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Surveillance Requirement 4.7.B.4 for all other Type B and C penetrations, the combined leakage rate is $\leq 0.60\ L_a$.

2. At least once per 31 days by verifying that all primary containment penetrations[b] not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except for valves that are open under administrative control as permitted by Specification 3.7.D.

3. By verifying each primary containment air lock is in compliance with the requirements of Specification 3.7.C.

4. By verifying the suppression chamber is in compliance with the requirements of Specification 3.7.K.

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a See Special Test Exception 3.12.A.

b Except valves, blind flanges, and deactivated automatic valves which are located inside the containment, and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed when the primary containment has not been de-inerted since the last verification or more often than once per 92 days.

3.7 - LIMITING CONDITIONS FOR OPERATION

B. Primary Containment Leakage

Primary containment leakage rates shall be limited to:

1. An overall integrated leakage rate of $\leq L_a$ which is defined as 1.6 percent by weight of the containment air per 24 hours at P_a (48 psig).

2. A combined leakage rate of $\leq 0.60\ L_a$ for all primary containment penetrations, except[a] for main steam line isolation valves, subject to Type B and C tests when pressurized to P_a (48 psig).

3. ≤ 11.5 scfh for any one main steam line isolation valve when tested at P_t (25 psig)[a].

APPLICABILITY:

When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.7.A.

ACTION:

With the measured combined leakage rate for all primary containment penetrations subject to Type B and C tests $>0.60\ L_a$, restore the combined leakage rate to $\leq 0.60\ L_a$, within 1 hour. Otherwise, be in HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

B. Primary Containment Leakage

The primary containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria, methods and provisions specified in Appendix J of 10CFR Part 50:

1. Three Type A overall integrated containment leakage rate tests shall be conducted at approximately equal intervals during shutdown at $\geq P_a$ (48 psig) during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

2. If the results of any periodic Type A test are $>0.75\ L_a$, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If the results of two consecutive Type A tests are $>0.75\ L_a$, a Type A test shall be performed at least every 18 months until the results of two consecutive Type A tests are $\leq 0.75\ L_a$, at which time the above test schedule may be resumed.

3. The accuracy of each Type A test shall be verified by a supplemental test which:

   a. Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within $0.25\ L_a$.

a Exemption from Appendix J to 10CFR Part 50.

3.7 - LIMITING CONDITIONS FOR OPERATION

4.7 - SURVEILLANCE REQUIREMENTS

b. Has duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

c. Requires the quantity of gas to be bled from the containment during the supplemental test to be equivalent to at least 25% of the total measured leakage at $\geq P_a$ (48 psig).

4. Type B and C tests shall be conducted with gas at $\geq P_a$ (48 psig) at intervals no greater than 24 months except for tests involving:

   a. Air locks which shall be leak tested in accordance with Surveillance Requirement 4.7.C,

   b. Main steam line isolation valves[a] which shall be leak tested at $\geq P_t$ (25 psig)[a] at least once per 18 months, and

   c. Bolted double-gasketed seals which shall be leak tested at $\geq P_a$ (48 psig) following each closure of the seal and at least once every 18 months.

   The provisions of Specification 4.0.B are not applicable to the 24 month surveillance intervals.

a Exemption from Appendix J to 10CFR Part 50.

3.7 - LIMITING CONDITIONS FOR OPERATION

C. Primary Containment Air Locks

Each primary containment air lock shall be OPERABLE with:

1. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

2. An overall air lock leakage rate of $\leq 0.05\ L_a$ at P_a (48 psig).

APPLICABILITY:

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

ACTION:

1. With one primary containment air lock door inoperable:

   a. Maintain at least the OPERABLE air lock door closed[b] and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.

   b. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed[b] at least once per 31 days.

4.7 - SURVEILLANCE REQUIREMENTS

C. Primary Containment Air Locks

Each primary containment air lock shall be demonstrated OPERABLE:

1. By conducting an overall air lock leakage test at $\geq P_a$ (48 psig) and by verifying that the overall air lock leakage rate is within its limit:

   a. Within 72 hours of air lock opening when PRIMARY CONTAINMENT INTEGRITY is required, except when the air lock is being used for multiple entries, then at least once per 72 hours.

   b. At least once per 6 months[c], and

   c. Prior to establishing PRIMARY CONTAINMENT INTEGRITY following air lock opening.

2. Concurrent with each overall air lock leakage test conducted prior to establishing PRIMARY CONTAINMENT INTEGRITY, by verifying that only one door in each air lock can be opened at a time.

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a See Special Test Exception 3.12.A.

b Except during entry through an OPERABLE door to repair an inoperable door or to facilitate the removal of personnel for a cumulative time not to exceed one hour per year.

c The provisions of Specification 4.0.B are not applicable.

3.7 - LIMITING CONDITIONS FOR OPERATION

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

2. With the primary containment air lock interlock mechanism inoperable, restore the air lock interlock mechanism to OPERABLE status within 24 hours, or lock at least one air lock door closed and verify that the door is locked closed at least once per 31 days. Personnel entry and exit through the airlock is permitted provided one OPERABLE air lock door remains locked closed at all times and an individual is dedicated to assure that both air lock doors are not open simultaneously.

3. With the primary containment air lock inoperable, except as a result of an inoperable air lock door or interlock mechanism, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

3.7 - LIMITING CONDITIONS FOR OPERATION

D. Primary Containment Isolation Valves

Each primary containment isolation valve and reactor instrumentation excess flow check valve shall be OPERABLE[a].

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With one or more of the primary containment isolation valve(s)[b] inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 4 hours either:

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

   b. Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolated position[a], or

   c. Isolate each affected penetration by use of at least one closed manual valve or blind flange[a].

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

4.7 - SURVEILLANCE REQUIREMENTS

D. Primary Containment Isolation Valves

1. Each power-operated or automatic primary containment isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair, or replacement work is performed on the valve or its associated actuator, control, or power circuit by performance of a cycling test and verification of isolation time.

2. Each power-operated or automatic primary containment isolation valve required to close on an isolation signal, except traversing in-core probe system explosive isolation valves, shall be demonstrated OPERABLE at least once per 18 months by verifying that on a containment isolation test signal each automatic isolation valve actuates to its isolation position.

3. The isolation time of each power-operated or automatic primary containment isolation valve shall be determined to be within its limit when tested pursuant to Specification 4.0.E.

4. Each reactor instrumentation line excess flow check valve which fulfills a primary containment isolation function shall be demonstrated OPERABLE at least once per 18 months by verifying that the valve checks flow.

5. Each traversing in-core probe system explosive isolation valve shall be demonstrated OPERABLE:

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a Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

b Except main steam isolation valves (MSIVs). Required actions for inoperable MSIVs are provided in Specification 3.6.M.

3.7 - LIMITING CONDITIONS FOR OPERATION

2. With one or more reactor instrumentation line excess flow check valves inoperable, operation may continue and the provisions of Specification 3.0.C are not applicable, provided that within 4 hours either:

   a. The inoperable valve is restored to OPERABLE status, or

   b. The instrument line is isolated and the associated instrument is declared inoperable.

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

4.7 - SURVEILLANCE REQUIREMENTS

a. At least once per 31 days by verifying the continuity of the explosive charge.

b. At least once per 18 months by removing at least one explosive squib from each explosive valve such that each explosive squib in each explosive valve will be tested at least once per 36 months, and initiating the removed explosive squib(s). The replacement charge for the exploded squib(s) shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No squib shall remain in use beyond the expiration of its shelf-life or operating life, as applicable.

3.7 - LIMITING CONDITIONS FOR OPERATION

E. Suppression Chamber - Drywell Vacuum Breakers

Nine suppression chamber - drywell vacuum breakers shall be OPERABLE and twelve suppression chamber - drywell vacuum breakers shall be closed.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With one or more of the required suppression chamber - drywell vacuum breakers inoperable for opening but known to be closed, restore at least nine vacuum breakers to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one suppression chamber - drywell vacuum breaker open, restore the open vacuum breaker to the closed position within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With one position indicator of any OPERABLE suppression chamber - drywell vacuum breaker inoperable, restore the inoperable position indicator to OPERABLE status within 14 days or visually verify the vacuum breaker to be closed at least once per 24 hours. Otherwise, declare the vacuum breaker inoperable.

4.7 - SURVEILLANCE REQUIREMENTS

E. Suppression Chamber - Drywell Vacuum Breakers

Each suppression chamber - drywell vacuum breaker shall be:

1. Verified closed at least once per 7 days.

2. Demonstrated OPERABLE:

   a. At least once per 31 days and within 12 hours after any discharge of steam to the suppression chamber from one or more main steam relief valve(s), by cycling each vacuum breaker through at least one complete cycle of full travel.

   b. At least once per 31 days by verifying both position indicator(s) OPERABLE by observing expected valve movement during the cycling test.

   c. At least once per 18 months by:

      1) Verifying the force required to open the vacuum breaker, from the closed position, to be ≤0.5 psid, and

      2) Verifying both position indicators OPERABLE by performance of a CHANNEL CALIBRATION.

      3) Verifying that each valve's position indicator is capable of detecting disk displacement of ≥0.0625 inches.

3.7 - LIMITING CONDITIONS FOR OPERATION

F. Reactor Building - Suppression Chamber Vacuum Breakers

All reactor building - suppression chamber vacuum breakers shall be OPERABLE and closed.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With one reactor building - suppression chamber vacuum breaker line inoperable for opening with both valves known to be closed, restore the inoperable vacuum breaker line to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one reactor building - suppression chamber vacuum breaker line otherwise inoperable, verify at least one vacuum breaker in the line to be closed within 2 hours and restore the open vacuum breaker to the closed position within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With the position indicator of the air operated reactor building - suppression chamber vacuum breaker inoperable, restore the inoperable position indicator to OPERABLE status within 14 days or verify the vacuum breaker to be closed at least once per 24 hours by an

4.7 - SURVEILLANCE REQUIREMENTS

F. Reactor Building - Suppression Chamber Vacuum Breakers

Each reactor building - suppression chamber vacuum breaker shall be:

1. Verified closed at least once per 7 days.

2. Demonstrated OPERABLE:

   a. At least once per 92 days when tested pursuant to Specification 4.0.E by:

      1) Cycling the vacuum breaker through at least one test cycle.

      2) Verifying the air operated vacuum breaker position indicator OPERABLE by observing expected valve movement during the cycling test.

   b. At least once per 18 months by:

      1) Demonstrating that the force required to open each vacuum breaker does not exceed the equivalent of 0.5 psid.

      2) Verifying the air operated vacuum breaker position indicator OPERABLE by performance of a CHANNEL CALIBRATION.

3.7 - LIMITING CONDITIONS FOR OPERATION

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

4.7 - SURVEILLANCE REQUIREMENTS

3.7 - LIMITING CONDITIONS FOR OPERATION

G. Drywell Internal Pressure

The drywell internal pressure shall not exceed +1.5 psig[a].

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With the drywell internal pressure <1.0 psig during the applicable time period for OPERATIONAL MODE 1, restore the internal pressure to above the low pressure limit within 24 hours or reduce THERMAL POWER to <15% RATED THERMAL POWER within the next 8 hours.

2. With the drywell internal pressure otherwise outside of the specified limits, restore the internal pressure to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

G. Drywell Internal Pressure

The drywell internal pressure shall be determined to be within the limits at least once per 12 hours.

a In OPERATIONAL MODE 1, during the time period beginning within 24 hours after THERMAL POWER is >15% of RATED THERMAL POWER following startup, and ending within 24 hours prior to reducing THERMAL POWER to <15% of RATED THERMAL POWER preliminary to a scheduled reactor shutdown, the drywell internal pressure shall also be maintained ≥1.0 psig (except for up to 4 hours for required surveillance which reduces the differential pressure.)

3.7 - LIMITING CONDITIONS FOR OPERATION

H. Drywell - Suppression Chamber Differential Pressure

Differential pressure between the drywell and the suppression chamber shall be ≥1.0 psid[a].

APPLICABILITY:

OPERATIONAL MODE 1, during the time period:

1. Beginning within 24 hours after THERMAL POWER is >15% of RATED THERMAL POWER following startup, and

2. Ending within 24 hours prior to reducing THERMAL POWER to <15% of RATED THERMAL POWER preliminary to a scheduled reactor shutdown.

ACTION:

1. With the drywell - suppression chamber differential pressure less than the above limit, restore the required differential pressure within 24 hours or reduce THERMAL POWER to <15% RATED THERMAL POWER within the next 8 hours.

2. With the drywell - suppression chamber differential pressure instrumentation CHANNEL inoperable, restore the inoperable CHANNEL to OPERABLE status within 30 days or reduce THERMAL POWER to <15% RATED THERMAL POWER within the next 8 hours.

4.7 - SURVEILLANCE REQUIREMENTS

H. Drywell - Suppression Chamber Differential Pressure

1. The drywell - suppression chamber differential pressure shall be demonstrated to be within limits by verifying the differential pressure at least once per 12 hours.

2. At least one drywell - suppression chamber differential pressure instrumentation CHANNEL, and at least one drywell pressure and one suppression chamber pressure instrumentation CHANNEL shall be demonstrated OPERABLE by performance of a:

   a. CHANNEL CHECK at least once per 24 hours,

   b. THIS ITEM INTENTIONALLY LEFT BLANK

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a Except for up to 4 hours for required surveillance which reduces the differential pressure.

3.7 - LIMITING CONDITIONS FOR OPERATION

3. With the drywell and/or suppression chamber pressure instrumentation CHANNEL(s) inoperable, restore the inoperable CHANNEL(s) to OPERABLE status within 30 days or reduce THERMAL POWER to <15% RATED THERMAL POWER within the next 8 hours.

4. With the drywell - suppression chamber differential pressure instrumentation CHANNEL inoperable and with insufficient drywell and suppression chamber pressure instrumentation CHANNEL(s) OPERABLE to determine drywell - suppression chamber differential pressure, restore either the drywell - suppression chamber differential pressure instrumentation CHANNEL or sufficient drywell and suppression chamber pressure instrumentation CHANNEL(s) to determine drywell - suppression chamber differential pressure to OPERABLE status within 8 hours or reduce THERMAL POWER to <15% RATED THERMAL POWER within the next 8 hours.

4.7 - SURVEILLANCE REQUIREMENTS

3.7 - LIMITING CONDITIONS FOR OPERATION

I. Primary Containment Nitrogen System

The primary containment nitrogen system shall be OPERABLE with:

1. An OPERABLE inerting flow path, and

2. An OPERABLE make-up flow path.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

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4.7 - SURVEILLANCE REQUIREMENTS

I. Primary Containment Nitrogen System

The primary containment nitrogen system shall be demonstrated to be OPERABLE at least once per 31 days by verifying that:

1. The liquid nitrogen storage tank level is ≥70 inches, and

2. Each valve, manual, power operated or automatic, in the flow path not locked, sealed, or otherwise secured in position, is in its correct position.

3.7 - LIMITING CONDITIONS FOR OPERATION

J. Primary Containment Oxygen Concentration

The suppression chamber and drywell atmosphere oxygen concentration shall be <4% by volume.

APPLICABILITY:

OPERATIONAL MODE 1, during the time period:

1. Beginning within 24 hours after THERMAL POWER is >15% of RATED THERMAL POWER following startup, and

2. Ending within 24 hours prior to reducing THERMAL POWER to <15% of RATED THERMAL POWER preliminary to a scheduled reactor shutdown.

ACTION:

With the drywell and/or suppression chamber oxygen concentration exceeding the limit, restore the oxygen concentration to within the limit within 24 hours or reduce THERMAL POWER to <15% RATED THERMAL POWER within the next 8 hours.

4.7 - SURVEILLANCE REQUIREMENTS

J. Primary Containment Oxygen Concentration

The suppression chamber and drywell oxygen concentration shall be verified to be within the limit within 24 hours after THERMAL POWER is >15% of RATED THERMAL POWER and at least once per 7 days thereafter.

3.7 - LIMITING CONDITIONS FOR OPERATION

K. Suppression Chamber

The suppression chamber shall be OPERABLE with:

1. The suppression pool water level between 14' 6.5" and 14' 10.5",

2. A suppression pool maximum average water temperature of ≤95°F during OPERATIONAL MODE(s) 1 or 2, except that the maximum average temperature may be permitted to increase to:

   a. ≤105°F during testing which adds heat to the suppression pool.

   b. ≤110°F with THERMAL POWER ≤1% of RATED THERMAL POWER.

   c. ≤120°F with the main steam line isolation valves closed following a scram.

3. A total leakage between the suppression chamber and drywell of less than the equivalent leakage through a 1 inch diameter orifice at a differential pressure of 1.0 psid.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With the suppression pool water level outside the above limits, restore the water level to within the limits

4.7 - SURVEILLANCE REQUIREMENTS

K. Suppression Chamber

The suppression chamber shall be demonstrated OPERABLE:

1. By verifying the suppression pool water level to be within the limits at least once per 24 hours.

2. At least once per 24 hours by verifying the suppression pool average water temperature to be ≤95°F, except:

   a. At least once per 5 minutes during testing which adds heat to the suppression pool, by verifying the suppression pool average water temperature to be ≤105°F.

   b. At least once per hour when suppression pool average water temperature is ≥ 95°F, by verifying:

      1) Suppression pool average water temperature to be ≤ 110°F, and

      2) THERMAL POWER to be ≤ 1% of RATED THERMAL POWER after suppression pool average water temperature has exceeded 95 °F for more than 24 hours.

   c. At least once per 30 minutes with the main steam isolation valves closed following a scram and suppression pool average water temperature >95°F, by verifying suppression pool average water temperature to be ≤120°F.

3.7 - LIMITING CONDITIONS FOR OPERATION

within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. In OPERATIONAL MODE(s) 1 or 2 with the suppression pool average water temperature >95°F, except as permitted above, restore the average temperature to ≤95°F within 24 hours or reduce THERMAL POWER to ≤1% RATED THERMAL POWER within the next 12 hours.

3. With the suppression pool average water temperature >105°F during testing which adds heat to the suppression pool, except as permitted above, stop all testing which adds heat to the suppression pool and restore the average temperature to ≤95°F within 24 hours or reduce THERMAL POWER to ≤1% RATED THERMAL POWER within the next 12 hours.

4. With the suppression pool average water temperature >110°F, immediately place the reactor mode switch in the Shutdown position and operate at least one low pressure coolant injection loop in the suppression pool cooling mode.

5. With the suppression pool average water temperature >120°F, depressurize the reactor pressure vessel to <150 psig (reactor steam dome pressure) within 12 hours.

4.7 - SURVEILLANCE REQUIREMENTS

3. By an external visual examination of the suppression chamber after main steam relief valve operation with the suppression pool average water temperature ≥160°F and reactor coolant system pressure >150 psig.

4. At least once per 18 months by a visual inspection of the accessible interior and exterior of the suppression chamber.

5. At least once per 18 months by conducting a drywell to suppression chamber bypass leak test at an initial differential pressure of 1.0 psid and verifying that the measured leakage is within the specified limit. If any drywell to suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the specified limit, a test shall be performed at least every 9 months until two consecutive tests meet the specified limit, at which time the 18 month test schedule may be resumed.

3.7 - LIMITING CONDITIONS FOR OPERATION

L. Suppression Chamber and Drywell Spray

The suppression chamber and drywell spray functions of the low pressure coolant injection (LPCI)/containment cooling system shall be OPERABLE with two independent loops, each loop consisting of:

1. One OPERABLE LPCI pump, and

2. An OPERABLE flow path capable of recirculating water from the suppression pool through a heat exchanger and the suppression chamber and drywell spray nozzles.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With one suppression chamber/drywell spray loop inoperable, restore the inoperable loop to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With both suppression chamber/drywell spray loops inoperable, restore at least one loop to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the next 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

L. Suppression Chamber and Drywell Spray

The suppression chamber and drywell spray functions of LPCI/containment cooling system shall be demonstrated OPERABLE:

1. At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

2. By performance of an air or smoke flow test of the drywell spray nozzles at least once per 5 years and verifying that each spray nozzle is unobstructed.

3.7 - LIMITING CONDITIONS FOR OPERATION

M. Suppression Pool Cooling

The suppression pool cooling function of the low pressure coolant injection (LPCI)/containment cooling system shall be OPERABLE with two independent loops, each loop consisting of:

1. One OPERABLE LPCI pump, and

2. An OPERABLE flow path capable of recirculating water from the suppression pool through a heat exchanger.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With one suppression pool cooling loop inoperable, restore the inoperable loop to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With both suppression pool cooling loops inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

M. Suppression Pool Cooling

The suppression pool cooling function of the LPCI/containment cooling system shall be demonstrated OPERABLE:

1. At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

2. By verifying that each of the required LPCI pumps develops the required recirculation flow through the heat exchanger and the suppression pool when tested pursuant to Specification 4.0.E.

3.7 - LIMITING CONDITIONS FOR OPERATION

N. SECONDARY CONTAINMENT INTEGRITY

SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY:

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

ACTION:

1. Without SECONDARY CONTAINMENT INTEGRITY in OPERATIONAL MODES(s) 1, 2 or 3, restore SECONDARY CONTAINMENT INTEGRITY within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. Without SECONDARY CONTAINMENT INTEGRITY in OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

4.7 - SURVEILLANCE REQUIREMENTS

N. SECONDARY CONTAINMENT INTEGRITY

SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

1. Verifying at least once per 24 hours that the pressure within the secondary containment is ≥0.25 inches of vacuum water gauge.

2. Verifying at least once per 31 days that:

   a. At least one door in each secondary containment air lock is closed.

   b. All secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic dampers secured in position.

3. At least once per 18 months by operating one standby gas treatment subsystem at a flow rate ≤4000 cfm for one hour and maintaining ≥0.25 inches of vacuum water gauge in the secondary containment.

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* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.7 - LIMITING CONDITIONS FOR OPERATION

O. Secondary Containment Automatic Isolation Dampers

Each secondary containment ventilation system automatic isolation damper shall be OPERABLE.

APPLICABILITY:

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

ACTION:

With one or more of the secondary containment ventilation system automatic isolation dampers inoperable, maintain at least one isolation damper OPERABLE in each affected penetration that is open and within 8 hours either:

1. Restore the inoperable damper(s) to OPERABLE status, or

2. Isolate each affected penetration by use of at least one deactivated automatic damper secured in the isolation position, or

3. Isolate each affected penetration by use of at least one closed manual valve or blind flange.

Otherwise, in OPERATIONAL MODE(s) 1, 2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Otherwise, in OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE

4.7 - SURVEILLANCE REQUIREMENTS

O. Secondary Containment Automatic Isolation Dampers

Each secondary containment ventilation system automatic isolation damper shall be demonstrated OPERABLE:

1. Prior to returning the damper to service after maintenance, repair, or replacement work is performed on the valve/damper or its associated actuator, control, or power circuit by performance of a cycling test.

2. At least once per 18 months by verifying that on an isolation test signal each automatic isolation damper actuates to its isolation position.

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* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.7 - LIMITING CONDITIONS FOR OPERATION

ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

4.7 - SURVEILLANCE REQUIREMENTS

3.7 - LIMITING CONDITIONS FOR OPERATION

P. Standby Gas Treatment System

Two independent standby gas treatment subsystems shall be OPERABLE, each with an OPERABLE diesel generator power source.

APPLICABILITY:

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

ACTION:

1. With one standby gas treatment subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or:

   a. In OPERATIONAL MODE(s) 1,2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   b. In OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

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4.7 - SURVEILLANCE REQUIREMENTS

P. Standby Gas Treatment System

Each standby gas treatment subsystem shall be demonstrated OPERABLE:

1. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters operating.

2. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the subsystem by:

   a. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of <1% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 4000 cfm ±10%.

   b. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM-D-3803-89, for a methyl iodide penetration of <10%, when tested at 30°C and 70% relative humidity; and

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* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.7 - LIMITING CONDITIONS FOR OPERATION

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4.7 - SURVEILLANCE REQUIREMENTS

c. Verifying a subsystem flow rate of 4000 cfm ±10% during system operation when tested in accordance with ANSI N510-1980.

3. After every # hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM-D-3803-89, for a methyl iodide penetration of <10%, when tested at 30°C and 70% relative humidity.

4. At least once per 18 months by:

   a. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is <6 inches water gauge while operating the filter train at a flow rate of 4000 cfm ±10%.

   b. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

      1) Manual initiation from the control room, and

      2) Simulated automatic initiation signal.

   c. Verifying that the heaters dissipate 30 ±3 kw when tested in accordance with ANSI N510-1980. This reading shall include the appropriate correction for variations from 480 volts at the bus.

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3.7 - LIMITING CONDITIONS FOR OPERATION

4. With both standby gas treatment subsystems otherwise inoperable in OPERATIONAL MODE(s) 1,2 or 3, restore at least one subsystem to OPERABLE status within one hour, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

5. With both standby gas treatment subsystems inoperable in OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

4.7 - SURVEILLANCE REQUIREMENTS

5. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the in-place penetration and leakage testing acceptance criteria of <1% in accordance with ANSI N510-1980 while operating the system at a flow rate of 4000 cfm ±10%.

6. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the in-place penetration and leakage testing acceptance criteria of <1% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 4000 cfm ±10%.

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* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3/4.7.A PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10CFR Part 100 during accident conditions.

3/4.7.B Primary Containment Leakage

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 48 psig, P_a. As an added conservatism, the measured overall integrated leakage rate, L_a, is further limited to $\leq 0.75\ L_a$ during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests. Periodic testing of the containment boundary is required to verify the allowable leakage rates are met. Generally, these tests are conducted while shutdown and the leakage rates must be verified as acceptable prior to establishing containment integrity. Type B and C tests may, however, be conducted at power and be found to exceed the limit while in the applicable OPERATIONAL MODE(s). A short time frame, consistent with the allowed outage time for PRIMARY CONTAINMENT INTEGRITY, is provided to restore the leakage to within its limits. If the leakage is tested individually for each valve in a penetration, closing and locking the other containment isolation valve with an acceptable leakage rate restores PRIMARY CONTAINMENT INTEGRITY. Locking the valve with an acceptable leakage rate is required to assure that the leakage rate is not exceeded due to a single failure that could cause the valve to be re-opened.

The main steam line isolation valves are tested at lower pressures, per an approved exemption, but the leakage rate is included in the Type B and C test totals. The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10CFR Part 50 with the exception of approved exemptions. (Ref: Exemption Request Approval, Mr. D. G. Eisenhut (NRC) to Mr. L. DelGeorge (CECo) dated June 25, 1982.)

3/4.7.C Primary Containment Air Locks

The limitations on closure and leakage for the primary containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the primary containment leakage rate given in Specifications 3.7.A. and 3.7.B. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation.

3/4.7.D Primary Containment Isolation Valves

The OPERABILITY of the primary containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

The containment is also penetrated by a large number of small diameter instrument lines which contact the primary coolant system. A program for periodic testing and examination of the flow check valves in these lines is performed by blowing down the instrument line during a vessel hydro and observing conditions which verify that the flow check valve is operable, e.g., a distinctive 'click' when the poppet valve seats, or an instrumentation high flow that quickly reduces to a slight trickle.

3/4.7.E Suppression Chamber - Drywell Vacuum Breakers

The function of the suppression chamber to drywell vacuum breakers is to relieve vacuum in the drywell. These internal vacuum breakers allow air and steam flow from the suppression chamber to the drywell when the drywell is at a negative pressure with respect to the suppression chamber. Each vacuum breaker is a self-actuating valve, similar to a check valve.

The safety analysis assumes that the internal vacuum breakers are closed initially and are fully open at a differential pressure of 0.5 psid. Additionally, three of these internal vacuum breakers are assumed to fail in a closed position. The results of the analyses show that the design pressure is not exceeded even under the worst case accident scenario.

The vacuum breakers between the suppression chamber and the drywell must not be inoperable in the open position since this would allow bypassing of the suppression pool in case of an accident. There is a sufficient number of valves so that operation may continue for a limited time with up to three vacuum breakers inoperable in the closed position.

Each suppression chamber to drywell vacuum breaker is fitted with a redundant pair of position switches which provide signals of disk position to panel mounted indicators and annunciate an alarm in the control room if the disk is open more than allowable. The alarm systems meet the intent of IEEE-279 standards.

3/4.7.F Reactor Building -Suppression Chamber Vacuum Breakers

The function of the reactor building to suppression chamber vacuum breakers is to relieve vacuum when the suppression chamber atmosphere depressurizes below reactor building pressure. If the drywell depressurizes below reactor building pressure, the negative differential pressure is

mitigated by flow through the reactor building to suppression chamber vacuum breakers and through the suppression chamber to drywell vacuum breakers. The reactor building to suppression chamber vacuum breakers include both an air operated valve and a check valve in each line. However, position indication is only provided on the air operated valve. These lines and valves are sized on the basis of the air flow from the secondary containment that is required to mitigate the depressurization transient and limit the maximum negative drywell pressure to within design limits. The maximum depressurization rate is a function of the drywell spray flow rate and temperature and the assumed initial conditions of the drywell atmosphere. The safety analyses assume the external vacuum breakers to be closed initially and to be fully open at 1.0 psid. Both vacuum breakers are periodically demonstrated to open at the required pressure differential. For the air operated vacuum breaker, this demonstration is essentially a CHANNEL CALIBRATION of the logic system. Additionally, of the two reactor building to suppression chamber vacuum breaker lines, one is assumed to fail in a closed position to satisfy the single active failure criterion.

3/4.7.G Drywell Internal Pressure

The limitations on drywell internal pressure ensure that the containment peak pressure does not exceed the design pressure during the Design Basis Accident (DBA). The upper limit for initial positive containment pressure will limit the total post accident design basis pressure to approximately 48 psig which is less than the design pressure and is consistent with the safety analysis. The maximum pressure, and the minimum pressure above 15% RATED THERMAL POWER, is also based on assumptions for post-accident hydrodynamic loading analysis. A short period is allowed to conduct testing, e.g. HPCI, vacuum breaker and relief valve testing, which temporarily reduces the drywell pressure below this minimum.

3/4.7.H Drywell-Suppression Chamber Differential Pressure

The toroidal-shaped suppression chamber, which contains the suppression pool is connected to the drywell by eight main vent pipes. The main vent pipes exhaust into a vent header, from which downcomer pipes extend into the suppression pool. During a loss-of-coolant accident (LOCA), the increasing drywell pressure will force the water leg in the downcomer pipes into the suppression pool at substantial velocities as the blowdown phase of the event begins. The length of the water leg has a significant effect on the resultant primary containment pressures and loads.

The purpose of maintaining the drywell at a slightly higher pressure with respect to the suppression chamber is to minimize the drywell pressure increase necessary to clear the downcomer pipes to commence condensation of steam in the suppression pool and to minimize the mass of the accelerated water leg. This reduces the hydrodynamic loads on the torus during the LOCA blowdown. Initial drywell-to-suppression-chamber differential pressure affects both the dynamic pool loads on the suppression chamber and the peak drywell pressure during downcomer pipe clearing during a Design Basis Accident. Drywell-to-suppression-chamber differential pressure must be maintained within the specified limits so that the safety analysis remains valid. However, a short period is allowed to conduct testing, e.g. HPCI, vacuum breaker and relief valve testing,

which temporarily increases the suppression chamber pressure and reduces the differential pressure. Only one direct suppression chamber to drywell differential pressure instrumentation CHANNEL is provided. However, any pair of the redundant drywell and suppression chamber pressure instrumentation CHANNEL(s) are sufficient to determine the differential pressure.

3/4.7.I Primary Containment Nitrogen System

The nitrogen system functions to maintain oxygen concentrations within the primary containment at or below the explosive levels. To ensure that a combustible gas mixture does not occur, oxygen concentration is kept below 4.0 volume percent. The system operates in conjunction with emergency operating procedures that are used to reduce primary containment pressure periodically during system operation. This combination results in a feed-and-bleed approach to maintaining hydrogen and/or oxygen concentrations below combustible levels. Sufficient liquid nitrogen is maintained to provide approximately a seven day supply to allow for establishing an additional nitrogen supply following a LOCA.

3/4.7.J Primary Containment Oxygen Concentration

All nuclear reactors must be designed to withstand events that generate hydrogen either due to the zirconium metal-water reaction in the core or due to radiolysis. The primary method to control hydrogen is to inert the primary containment. With the primary containment inerted, that is, oxygen concentration less than 4.0 volume percent, a combustible mixture cannot be present in the primary containment for any hydrogen concentration. The Design Basis Accident (DBA) loss-of-coolant accident (LOCA) analysis assumes that the primary containment is inerted when the DBA occurs. Thus, the hydrogen assumed to be released to the primary containment as a result of a metal-water reaction in the reactor core will not produce combustible gas mixtures in the primary containment.

The primary containment oxygen concentration must be within the specified limit when primary containment is inerted, except as allowed by the relaxations during startup and shutdown. The primary containment must be inert in OPERATIONAL MODE 1, since this is the condition with the highest probability of an event that could produce hydrogen. Inerting the primary containment is an operational problem because it prevents containment access without an appropriate breathing apparatus. Therefore, the primary containment is inerted as late as possible in the plant startup and de-inerted as soon as possible in the plant shutdown. As long as reactor power is below 15% of RATED THERMAL POWER, the potential for an event that generates significant hydrogen is low and the primary containment does not need to be inert. Furthermore, the probability of an event that generates hydrogen occurring within the first 24 hours of a reactor startup or within the last 24 hours before a shutdown is low enough that these windows, when the primary containment is not inerted, are also justified. The 24 hour time frame is a reasonable amount of time to allow plant personnel to perform inerting or de-inerting.

3/4.7.K Suppression Chamber

The specifications of this section ensure that the primary containment pressure will not exceed the design pressure during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from ~1000 psig. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss of coolant accident, the pressure of the liquid and gas must not exceed the suppression chamber maximum pressure. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

An allowable bypass area between the primary containment and the drywell and suppression chamber is identified based on analysis considering primary system break area, suppression chamber effectiveness, and containment design pressure. Analyses show that the maximum allowable bypass area is equivalent to all vacuum breakers open the equivalent of 1/16 inch at all points along the seal surface of the disk.

Using the minimum or maximum water levels given in this specification (as measured from the bottom of the suppression chamber), primary containment maximum pressure following a design basis accident is approximately 48 psig, which is below the design pressure. The maximum water level results in a downcomer submergence of 4 feet and the minimum level results in a submergence approximately 4 inches less. If it becomes necessary to make the suppression chamber inoperable, it is done in accordance with the requirements in Specification 3.5.C.

Because of the large volume and thermal capacity of the suppression pool, the level and temperature normally change very slowly and monitoring these parameters daily is sufficient to establish any trend. By requiring the suppression pool temperature to be more frequently monitored during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

Under full power operating conditions, blowdown from an initial suppression chamber water temperature of 95°F results in a water temperature of approximately 145°F immediately following blowdown which is low enough to provide complete condensation via T-quencher devices. At this temperature and atmospheric pressure, the available net positive suction head exceeds that required by the emergency core cooling system pumps, thus there is no dependency on containment overpressure during the accident injection phase.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained sufficiently low during any period of safety relief valve operation for T-quencher devices. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings. In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a safety or relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety or relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety or relief valve to assure mixing and uniformity of energy insertion to the pool.

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping. The maintenance of a drywell-suppression chamber differential pressure and a suppression chamber water level corresponding to a downcomer submergence range of 3.67 to 4.00 feet will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

3/4.7.L Suppression Chamber and Drywell Spray

Following a Design Basis Accident (DBA), the suppression chamber spray function of the low pressure coolant injection (LPCI)/containment cooling system removes heat from the suppression chamber air space and condenses steam. The suppression chamber is designed to absorb the sudden input of heat from the primary system from a DBA or a rapid depressurization of the reactor pressure vessel through safety or relief valves. There is one 100% capacity containment spray header inside the suppression chamber. Periodic operation of the suppression chamber and drywell sprays may also be used following a DBA to assist the natural convection and diffusion mixing of hydrogen and oxygen when other ECCS requirements are met and oxygen concentration exceeds 4%. Since the spray system is a function of the LPCI/containment cooling system, the loops will not be aligned for the spray function during normal operation, but all components required to operate for proper alignment must be OPERABLE.

3/4.7.M Suppression Pool Cooling

Following an accident, the suppression pool cooling function of the LPCI/containment cooling system removes heat that the suppression pool absorbs from the primary system and, in the long term, continues to absorb residual heat generated by fuel in the reactor core. Each of the suppression pool cooling loops consists of a pump and heat exchanger. Following a loss of coolant accident (LOCA), the plant operators can realign the valves in these two loops to draw water from the suppression pool, pump it through the shell side of the exchangers, and discharge it back to the suppression pool via the full flow test lines. At the same time, containment cooling service

water (CCSW) is pumped through the tube side of the exchangers to exchange heat to the external heat sink.

3/4.7.N SECONDARY CONTAINMENT INTEGRITY

The function of the secondary containment is to isolate and contain fission products that escape from primary containment following a Design Basis Accident (DBA), to confine the postulated release of radioactive material within the requirements of 10CFR Part 100, and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside of primary containment. The reactor building and associated structures provide secondary containment during normal operation when the drywell is sealed and in service. At other times the drywell may be open and, when required, secondary containment integrity is specified. There are two principal accidents for which credit is taken for secondary containment OPERABILITY. These are a LOCA and fuel-handling accident inside secondary containment. The secondary containment performs no active function in response to each of these limiting events; however, its leak tightness is required to limit offsite radiation doses to below those required by 10CFR Part 100. Maintaining secondary containment OPERABLE ensures that the release of radioactive materials from the primary containment is restricted to those leakage paths and associated leakage rates assumed in the accident analysis and that fission products entrapped within the secondary containment structure will be treated prior to discharge to the environment. Establishing and maintaining a vacuum in the reactor building with the standby gas treatment system during testing, along with the surveillance of the doors, hatches, dampers and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment. This surveillance is normally conducted during periods of calm winds (<5 mph), but may be conducted under higher wind conditions with appropriate application of correction factors.

3/4.7.O Secondary Containment Automatic Isolation Dampers

The function of the secondary containment ventilation system automatic isolation dampers, in combination with other accident-mitigation systems, is to limit fission-product release during and following postulated Design Basis Accidents (DBA) such that offsite radiation exposures are maintained within the requirements of 10CFR Part 100. Secondary containment isolation ensures that fission products that escape from primary containment following a DBA, or which are released during certain operations when primary containment is not required, or take place outside primary containment, are maintained within applicable limits. The OPERABILITY requirements for the secondary containment ventilation system isolation dampers help ensure that adequate secondary containment leak tightness is maintained during and after an accident by minimizing potential paths to the environment.

3/4.7.P Standby Gas Treatment System

The standby gas treatment system (SBGT) is required to ensure that radioactive materials that leak from the primary containment into the secondary containment following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment. This system reduces the potential releases of radioactive material, principally iodine, to within values specified in 10CFR Part 100.

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the main chimney during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environment. One standby gas treatment fan is designed to automatically start upon secondary containment isolation and to maintain the reactor building pressure to approximately a negative ¼ inch water gauge pressure; all leakage should be in-leakage. Should the fan fail to start, the redundant alternate fan and filter subsystem is designed to start automatically.

The OPERABILITY of the standby gas treatment system reduces the potential release of radioactive material, principally iodine, following a design basis accident. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses. Periodic operation of the system with the heaters is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.

Since the standby gas treatment subsystems are shared by both units, one subsystem is powered by the unit diesel generator power source of each unit. This unique arrangement requires that special allowed out-of-service times be provided for the combinations of subsystem and diesel generator power source inoperability that may occur. For example, if conducting the alternate offsite power source cross-tie surveillance were to require the inoperability of both unit diesel generator power sources, neither of the standby gas treatment subsystems would have an OPERABLE diesel generator power source and the appropriate ACTION would have to be entered.

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

COMMONWEALTH EDISON COMPANY

AND

MIDAMERICAN ENERGY COMPANY

DOCKET NO. 50-254

QUAD CITIES NUCLEAR POWER STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 165
License No. DPR-29

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated September 17, 1993, as supplemented by letter dated July 20, 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:

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P PDR

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 165 , 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: November 27, 1995

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

COMMONWEALTH EDISON COMPANY

AND

MIDAMERICAN ENERGY COMPANY

DOCKET NO. 50-265

QUAD CITIES NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 161
License No. DPR-30

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated September 17, 1993, as supplemented by letter dated July 20, 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. 161 , 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: November 27, 1995

ATTACHMENT TO LICENSE AMENDMENT NOS. 165 AND 161

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
3.7/4.7-1	3.7/4.7-1	3/4.7-1
3.7/4.7-2	3.7/4.7-2	3/4.7-2
3.7/4.7-3	3.7/4.7-3	3/4.7-3
3.7/4.7-4	3.7/4.7-4	3/4.7-4
3.7/4.7-5	3.7/4.7-5	3/4.7-5
3.7/4.7-6	3.7/4.7-6	3/4.7-6
3.7/4.7-7	3.7/4.7-6a	3/4.7-7
3.7/4.7-8	3.7/4.7-6b	3/4.7-8
3.7/4.7-9	3.7/4.7-6c	3/4.7-9
3.7/4.7-10	3.7/4.7-6d	3/4.7-10
3.7/4.7-11	3.7/4.7-7	3/4.7-11
3.7/4.7-11a	3.7/4.7-7a	3/4.7-12
3.7/4.7-11b	3.7/4.7-7b	3/4.7-13
3.7/4.7-11c	3.7/4.7-7c	3/4.7-14
3.7/4.7-11d	3.7/4.7-8	3/4.7-15
3.7/4.7-12	3.7/4.7-9	3/4.7-16
3.7/4.7-13	3.7/4.7-10	3/4.7-17
3.7/4.7-14	3.7/4.7-10a	3/4.7-18
3.7/4.7-15	3.7/4.7-11	3/4.7-19
3.7/4.7-16	3.7/4.7-11a	3/4.7-30
3.7/4.7-17	3.7/4.7-12	3/4.7-21
3.7/4.7-18	3.7/4.7-13	3/4.7-22
3.7/4.7-19	3.7/4/7-14	3/4.7-23
3.7/4.7-20	3.7/4.7-15	3/4.7-24
3.7/4.7-21	3.7/4/7-16	3/4.7-25
3.7/4.7-21a	3.7/4/7-17	3/4.7-26
3.7/4.7-22	3.7/4.7-18	B 3/4.7-1
3.7/4.7-23	3.7/4.7-19	B 3/4.7-2
3.7/4.7-24	3.7/4.7-20	B 3/4.7-3
3.7/4.7-25	3.7/4.7-20a	B 3/4.7-4
3.7/4.7-26	3.7/4.7-20b	B 3/4.7-5
3.7/4.7-27	3.7/4.7-21	B 3/4.7-6
3.7/4.7-28	3.7/4.7-21a	B 3/4.7-7
3.7/4.7-29	3.7/4.7-22	B 3/4.7-8
3.7/4.7-30	3.7/4.7-23	---
3.7/4.7-31	---	---
3.7/4.7-32	---	---
3.7/4.7-33	---	---
3.7/4.7-34	---	---
3.7/4.7-35	---	---
3.7/4.7-36	---	---

UNIT 1 REMOVE	UNIT 2 REMOVE	INSERT
3.7/4.7-37	---	---
3.7/4.7-38	---	---
3.7/4.7-39	---	---
3.7/4.7-40	---	---

3.7 - LIMITING CONDITIONS FOR OPERATION

A. PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY:

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

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

A. PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

1. After each closing of each penetration subject to Type B testing, except the primary containment air locks, if opened following Type A or B test, by leak rate testing the seals with gas at $\geq P_a$ (48 psig), and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Surveillance Requirement 4.7.B.4 for all other Type B and C penetrations, the combined leakage rate is $\leq 0.60\ L_a$.

2. At least once per 31 days by verifying that all primary containment penetrations[b] not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except for valves that are open under administrative control as permitted by Specification 3.7.D.

3. By verifying each primary containment air lock is in compliance with the requirements of Specification 3.7.C.

4. By verifying the suppression chamber is in compliance with the requirements of Specification 3.7.K.

---

a See Special Test Exception 3.12.A.

b Except valves, blind flanges, and deactivated automatic valves which are located inside the containment, and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed when the primary containment has not been de-inerted since the last verification or more often than once per 92 days.

3.7 - LIMITING CONDITIONS FOR OPERATION

B. Primary Containment Leakage

Primary containment leakage rates shall be limited to:

1. An overall integrated leakage rate of $\leq L_a$ which is defined as 1.0 percent by weight of the containment air per 24 hours at P_a (48 psig).

2. A combined leakage rate of $\leq 0.60\ L_a$ for all primary containment penetrations, except[a] for main steam line isolation valves, subject to Type B and C tests when pressurized to P_a (48 psig).

3. ≤ 11.5 scfh for any one main steam line isolation valve when tested at P_t (25 psig)[a].

APPLICABILITY:

When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.7.A.

ACTION:

With the measured combined leakage rate for all primary containment penetrations subject to Type B and C tests $>0.60\ L_a$, restore the combined leakage rate to $\leq 0.60\ L_a$, within 1 hour. Otherwise, be in HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

B. Primary Containment Leakage

The primary containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria, methods and provisions specified in Appendix J of 10CFR Part 50:

1. Three Type A overall integrated containment leakage rate tests shall be conducted at approximately equal intervals during shutdown at $\geq P_a$ (48 psig) during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

2. If the results of any periodic Type A test are $>0.75\ L_a$, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If the results of two consecutive Type A tests are $>0.75\ L_a$, a Type A test shall be performed at least every 18 months until the results of two consecutive Type A tests are $\leq 0.75\ L_a$, at which time the above test schedule may be resumed.

3. The accuracy of each Type A test shall be verified by a supplemental test which:

   a. Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within $0.25\ L_a$.

a Exemption from Appendix J to 10CFR Part 50.

3.7 - LIMITING CONDITIONS FOR OPERATION

4.7 - SURVEILLANCE REQUIREMENTS

b. Has duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

c. Requires the quantity of gas to be bled from the containment during the supplemental test to be equivalent to at least 25% of the total measured leakage at $\geq P_a$ (48 psig).

4. Type B and C tests shall be conducted with gas at $\geq P_a$ (48 psig) at intervals no greater than 24 months except for tests involving:

a. Air locks which shall be leak tested in accordance with Surveillance Requirement 4.7.C,

b. Main steam line isolation valves[a] which shall be leak tested at $\geq P_t$ (25 psig)[a] at least once per 18 months, and

c. Bolted double-gasketed seals which shall be leak tested at $\geq P_a$ (48 psig) following each closure of the seal and at least once every 18 months.

The provisions of Specification 4.0.B are not applicable to the 24 month surveillance intervals.

a Exemption from Appendix J to 10CFR Part 50.

3.7 - LIMITING CONDITIONS FOR OPERATION

C. Primary Containment Air Locks

Each primary containment air lock shall be OPERABLE with:

1. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

2. An overall air lock leakage rate of $\leq 0.05\ L_a$ at P_a (48 psig).

APPLICABILITY:

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

ACTION:

1. With one primary containment air lock door inoperable:

   a. Maintain at least the OPERABLE air lock door closed[b] and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed.

   b. Operation may then continue until performance of the next required overall air lock leakage test provided that the OPERABLE air lock door is verified to be locked closed[b] at least once per 31 days.

4.7 - SURVEILLANCE REQUIREMENTS

C. Primary Containment Air Locks

Each primary containment air lock shall be demonstrated OPERABLE:

1. By conducting an overall air lock leakage test at $\geq P_a$ (48 psig) and by verifying that the overall air lock leakage rate is within its limit:

   a. Within 72 hours of air lock opening when PRIMARY CONTAINMENT INTEGRITY is required, except when the air lock is being used for multiple entries, then at least once per 72 hours.

   b. At least once per 6 months[c], and

   c. Prior to establishing PRIMARY CONTAINMENT INTEGRITY following air lock opening.

2. Concurrent with each overall air lock leakage test conducted prior to establishing PRIMARY CONTAINMENT INTEGRITY, by verifying that only one door in each air lock can be opened at a time.

---

a See Special Test Exception 3.12.A.

b Except during entry through an OPERABLE door to repair an inoperable door or to facilitate the removal of personnel for a cumulative time not to exceed one hour per year.

c The provisions of Specification 4.0.B are not applicable.

3.7 - LIMITING CONDITIONS FOR OPERATION

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

2. With the primary containment air lock interlock mechanism inoperable, restore the air lock interlock mechanism to OPERABLE status within 24 hours, or lock at least one air lock door closed and verify that the door is locked closed at least once per 31 days. Personnel entry and exit through the airlock is permitted provided one OPERABLE air lock door remains locked closed at all times and an individual is dedicated to assure that both air lock doors are not open simultaneously.

3. With the primary containment air lock inoperable, except as a result of an inoperable air lock door or interlock mechanism, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

3.7 - LIMITING CONDITIONS FOR OPERATION

D. Primary Containment Isolation Valves

Each primary containment isolation valve and reactor instrumentation excess flow check valve shall be OPERABLE[a].

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With one or more of the primary containment isolation valve(s)[b] inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 4 hours either:

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

   b. Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolated position[a], or

   c. Isolate each affected penetration by use of at least one closed manual valve or blind flange[a].

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

4.7 - SURVEILLANCE REQUIREMENTS

D. Primary Containment Isolation Valves

1. Each power-operated or automatic primary containment isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair, or replacement work is performed on the valve or its associated actuator, control, or power circuit by performance of a cycling test and verification of isolation time.

2. Each power-operated or automatic primary containment isolation valve required to close on an isolation signal, except traversing in-core probe system explosive isolation valves, shall be demonstrated OPERABLE at least once per 18 months by verifying that on a containment isolation test signal each automatic isolation valve actuates to its isolation position.

3. The isolation time of each power-operated or automatic primary containment isolation valve shall be determined to be within its limit when tested pursuant to Specification 4.0.E.

4. Each reactor instrumentation line excess flow check valve which fulfills a primary containment isolation function shall be demonstrated OPERABLE at least once per 18 months by verifying that the valve checks flow.

5. Each traversing in-core probe system explosive isolation valve shall be demonstrated OPERABLE:

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a Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

b Except main steam isolation valves (MSIVs). Required actions for inoperable MSIVs are provided in Specification 3.6.M.

3.7 - LIMITING CONDITIONS FOR OPERATION

2. With one or more reactor instrumentation line excess flow check valves inoperable, operation may continue and the provisions of Specification 3.0.C are not applicable, provided that within 4 hours either:

   a. The inoperable valve is restored to OPERABLE status, or

   b. The instrument line is isolated and the associated instrument is declared inoperable.

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

4.7 - SURVEILLANCE REQUIREMENTS

a. At least once per 31 days by verifying the continuity of the explosive charge.

b. At least once per 18 months by removing at least one explosive squib from each explosive valve such that each explosive squib in each explosive valve will be tested at least once per 36 months, and initiating the removed explosive squib(s). The replacement charge for the exploded squib(s) shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No squib shall remain in use beyond the expiration of its shelf-life or operating life, as applicable.

3.7 - LIMITING CONDITIONS FOR OPERATION

E. Suppression Chamber - Drywell Vacuum Breakers

Nine suppression chamber - drywell vacuum breakers shall be OPERABLE and twelve suppression chamber - drywell vacuum breakers shall be closed.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With one or more of the required suppression chamber - drywell vacuum breakers inoperable for opening but known to be closed, restore at least nine vacuum breakers to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one suppression chamber - drywell vacuum breaker open, restore the open vacuum breaker to the closed position within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With one position indicator of any OPERABLE suppression chamber - drywell vacuum breaker inoperable, verify the vacuum breaker(s) with the inoperable position indicator to be closed by conducting a test which demonstrates that the ΔP is

4.7 - SURVEILLANCE REQUIREMENTS

E. Suppression Chamber - Drywell Vacuum Breakers

Each suppression chamber - drywell vacuum breaker shall be:

1. Verified closed at least once per 7 days.

2. Demonstrated OPERABLE:

   a. At least once per 31 days and within 12 hours after any discharge of steam to the suppression chamber from one or more main steam relief valve(s), by cycling each vacuum breaker through at least one complete cycle of full travel.

   b. At least once per 31 days by verifying both position indicator(s) OPERABLE by observing expected valve movement during the cycling test.

   c. At least once per 18 months by:

      1) Verifying the force required to open the vacuum breaker, from the closed position, to be $\leq$0.5 psid, and

      2) Verifying both position indicators OPERABLE by performance of a CHANNEL CALIBRATION.

      3) Verifying that each valve's position indicator is capable of detecting disk displacement of $\geq$0.0625 inches.

3.7 - LIMITING CONDITIONS FOR OPERATION

maintained at greater than or equal to 0.5 psi for one hour without makeup within 24 hours and at least once per 15 days thereafter. Otherwise be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

3.7 - LIMITING CONDITIONS FOR OPERATION

F. Reactor Building - Suppression Chamber Vacuum Breakers

All reactor building - suppression chamber vacuum breakers shall be OPERABLE and closed.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With one reactor building - suppression chamber vacuum breaker line inoperable for opening with both valves known to be closed, restore the inoperable vacuum breaker line to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one reactor building - suppression chamber vacuum breaker line otherwise inoperable, verify at least one vacuum breaker in the line to be closed within 2 hours and restore the open vacuum breaker to the closed position within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With the position indicator of the air operated reactor building - suppression chamber vacuum breaker inoperable, restore the inoperable position indicator to OPERABLE status within 14 days or verify the vacuum breaker to be closed at least once per 24 hours by an

4.7 - SURVEILLANCE REQUIREMENTS

F. Reactor Building - Suppression Chamber Vacuum Breakers

Each reactor building - suppression chamber vacuum breaker shall be:

1. Verified closed at least once per 7 days.

2. Demonstrated OPERABLE:

   a. At least once per 92 days when tested pursuant to Specification 4.0.E by:

      1) Cycling the vacuum breaker through at least one test cycle.

      2) Verifying the air operated vacuum breaker position indicator OPERABLE by observing expected valve movement during the cycling test.

   b. At least once per 18 months by:

      1) Demonstrating that the force required to open each vacuum breaker does not exceed the equivalent of 0.5 psid.

      2) Verifying the air operated vacuum breaker position indicator OPERABLE by performance of a CHANNEL CALIBRATION.

3.7 - LIMITING CONDITIONS FOR OPERATION

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

4.7 - SURVEILLANCE REQUIREMENTS

3.7 - LIMITING CONDITIONS FOR OPERATION

G. Drywell Internal Pressure

The drywell internal pressure shall not exceed +1.5 psig[a].

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With the drywell internal pressure <1.0 psig during the applicable time period for OPERATIONAL MODE 1, restore the internal pressure to above the low pressure limit within 24 hours or reduce THERMAL POWER to <15% RATED THERMAL POWER within the next 8 hours.

2. With the drywell internal pressure otherwise outside of the specified limits, restore the internal pressure to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

G. Drywell Internal Pressure

The drywell internal pressure shall be determined to be within the limits at least once per 12 hours.

a In OPERATIONAL MODE 1, during the time period beginning within 24 hours after THERMAL POWER is >15% of RATED THERMAL POWER following startup, and ending within 24 hours prior to reducing THERMAL POWER to <15% of RATED THERMAL POWER preliminary to a scheduled reactor shutdown, the drywell internal pressure shall also be maintained ≥1.0 psig (except for up to 4 hours for required surveillance which reduces the differential pressure.)

3.7 - LIMITING CONDITIONS FOR OPERATION

H. Drywell - Suppression Chamber Differential Pressure

Differential pressure between the drywell and the suppression chamber shall be ≥1.0 psid[a].

APPLICABILITY:

OPERATIONAL MODE 1, during the time period:

1. Beginning within 24 hours after THERMAL POWER is >15% of RATED THERMAL POWER following startup, and

2. Ending within 24 hours prior to reducing THERMAL POWER to <15% of RATED THERMAL POWER preliminary to a scheduled reactor shutdown.

ACTION:

1. With the drywell - suppression chamber differential pressure less than the above limit, restore the required differential pressure within 24 hours or reduce THERMAL POWER to <15% RATED THERMAL POWER within the next 8 hours.

2. With the drywell - suppression chamber differential pressure instrumentation CHANNEL inoperable, restore the inoperable CHANNEL to OPERABLE status within 30 days or reduce THERMAL POWER to <15% RATED THERMAL POWER within the next 8 hours.

4.7 - SURVEILLANCE REQUIREMENTS

H. Drywell - Suppression Chamber Differential Pressure

1. The drywell - suppression chamber differential pressure shall be demonstrated to be within limits by verifying the differential pressure at least once per 12 hours.

2. At least one drywell - suppression chamber differential pressure instrumentation CHANNEL, and at least one drywell pressure and one suppression chamber pressure instrumentation CHANNEL shall be demonstrated OPERABLE by performance of a:

   a. CHANNEL CHECK at least once per 24 hours,

   b. CHANNEL CALIBRATION at least once per 18 months.

---

a Except for up to 4 hours for required surveillance which reduces the differential pressure.

3.7 - LIMITING CONDITIONS FOR OPERATION

3. With the drywell and/or suppression chamber pressure instrumentation CHANNEL(s) inoperable, restore the inoperable CHANNEL(s) to OPERABLE status within 30 days or reduce THERMAL POWER to <15% RATED THERMAL POWER within the next 8 hours.

4. With the drywell - suppression chamber differential pressure instrumentation CHANNEL inoperable and with insufficient drywell and suppression chamber pressure instrumentation CHANNEL(s) OPERABLE to determine drywell - suppression chamber differential pressure, restore either the drywell - suppression chamber differential pressure instrumentation CHANNEL or sufficient drywell and suppression chamber pressure instrumentation CHANNEL(s) to determine drywell - suppression chamber differential pressure to OPERABLE status within 8 hours or reduce THERMAL POWER to <15% RATED THERMAL POWER within the next 8 hours.

4.7 - SURVEILLANCE REQUIREMENTS

3.7 - LIMITING CONDITIONS FOR OPERATION

I. Primary Containment Nitrogen System

The primary containment nitrogen system shall be OPERABLE with:

1. An OPERABLE inerting flow path, and
2. An OPERABLE make-up flow path.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

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4.7 - SURVEILLANCE REQUIREMENTS

I. Primary Containment Nitrogen System

The primary containment nitrogen system shall be demonstrated to be OPERABLE at least once per 31 days by verifying that:

1. The liquid nitrogen storage tank level is ≥70 inches, and
2. Each valve, manual, power operated or automatic, in the flow path not locked, sealed, or otherwise secured in position, is in its correct position.

3.7 - LIMITING CONDITIONS FOR OPERATION

J. Primary Containment Oxygen Concentration

The suppression chamber and drywell atmosphere oxygen concentration shall be <4% by volume.

APPLICABILITY:

OPERATIONAL MODE 1, during the time period:

1. Beginning within 24 hours after THERMAL POWER is >15% of RATED THERMAL POWER following startup, and

2. Ending within 24 hours prior to reducing THERMAL POWER to <15% of RATED THERMAL POWER preliminary to a scheduled reactor shutdown.

ACTION:

With the drywell and/or suppression chamber oxygen concentration exceeding the limit, restore the oxygen concentration to within the limit within 24 hours or reduce THERMAL POWER to <15% RATED THERMAL POWER within the next 8 hours.

4.7 - SURVEILLANCE REQUIREMENTS

J. Primary Containment Oxygen Concentration

The suppression chamber and drywell oxygen concentration shall be verified to be within the limit within 24 hours after THERMAL POWER is >15% of RATED THERMAL POWER and at least once per 7 days thereafter.

3.7 - LIMITING CONDITIONS FOR OPERATION

K. Suppression Chamber

The suppression chamber shall be OPERABLE with:

1. The suppression pool water level between 14' 1" and 14' 5",

2. A suppression pool maximum average water temperature of ≤95°F during OPERATIONAL MODE(s) 1 or 2, except that the maximum average temperature may be permitted to increase to:

   a. ≤105°F during testing which adds heat to the suppression pool.

   b. ≤110°F with THERMAL POWER ≤1% of RATED THERMAL POWER.

   c. ≤120°F with the main steam line isolation valves closed following a scram.

3. A total leakage between the suppression chamber and drywell of less than the equivalent leakage through a 1 inch diameter orifice at a differential pressure of 1.0 psid.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With the suppression pool water level outside the above limits, restore the water level to within the limits

4.7 - SURVEILLANCE REQUIREMENTS

K. Suppression Chamber

The suppression chamber shall be demonstrated OPERABLE:

1. By verifying the suppression pool water level to be within the limits at least once per 24 hours.

2. At least once per 24 hours by verifying the suppression pool average water temperature to be ≤95°F, except:

   a. At least once per 5 minutes during testing which adds heat to the suppression pool, by verifying the suppression pool average water temperature to be ≤105°F.

   b. At least once per hour when suppression pool average water temperature is ≥95°F, by verifying:

      1) Suppression pool average water temperature to be ≤110°F, and

      2) THERMAL POWER to be ≤1% of RATED THERMAL POWER after suppression pool average water temperature has exceeded 95°F for more than 24 hours.

   c. At least once per 30 minutes with the main steam line isolation valves closed following a scram and suppression pool average water temperature >95°F, by verifying suppression pool average water temperature to be ≤120°F.

3.7 - LIMITING CONDITIONS FOR OPERATION

within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. In OPERATIONAL MODE(s) 1 or 2 with the suppression pool average water temperature >95°F, except as permitted above, restore the average temperature to ≤95°F within 24 hours or reduce THERMAL POWER to ≤1% RATED THERMAL POWER within the next 12 hours.

3. With the suppression pool average water temperature >105°F during testing which adds heat to the suppression pool, except as permitted above, stop all testing which adds heat to the suppression pool and restore the average temperature to ≤95°F within 24 hours or reduce THERMAL POWER to ≤1% RATED THERMAL POWER within the next 12 hours.

4. With the suppression pool average water temperature >110°F, immediately place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.

5. With the suppression pool average water temperature >120°F, depressurize the reactor pressure vessel to <150 psig (reactor steam dome pressure) within 12 hours.

4.7 - SURVEILLANCE REQUIREMENTS

3. By an external visual examination of the suppression chamber after main steam relief valve operation with the suppression pool average water temperature ≥160°F and reactor coolant system pressure >150 psig.

4. At least once per 18 months by a visual inspection of the accessible interior and exterior of the suppression chamber.

5. At least once per 18 months by conducting a drywell to suppression chamber bypass leak test at an initial differential pressure of 1.0 psid and verifying that the measured leakage is within the specified limit. If any drywell to suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the specified limit, a test shall be performed at least every 9 months until two consecutive tests meet the specified limit, at which time the 18 month test schedule may be resumed.

3.7 - LIMITING CONDITIONS FOR OPERATION

L. Suppression Chamber and Drywell Spray

The suppression chamber and drywell spray functions of the residual heat removal (RHR) system shall be OPERABLE with two independent subsystems, each subsystem consisting of:

1. One OPERABLE RHR pump, and

2. An OPERABLE flow path capable of recirculating water from the suppression pool through a heat exchanger and the suppression chamber and drywell spray nozzles.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With one suppression chamber/drywell spray subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With both suppression chamber/drywell spray subsystems inoperable, restore at least one subsystem to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN[a] within the next 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

L. Suppression Chamber and Drywell Spray

The suppression chamber and drywell spray functions of the RHR system shall be demonstrated OPERABLE:

1. At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

2. By performance of an air or smoke flow test of the drywell spray nozzles at least once per 5 years and verifying that each spray nozzle is unobstructed.

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a Whenever the two required RHR SDC mode subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

3.7 - LIMITING CONDITIONS FOR OPERATION

M. Suppression Pool Cooling

The suppression pool cooling function of the residual heat removal (RHR) system shall be OPERABLE with two independent subsystems, each subsystem consisting of:

1. One OPERABLE RHR pump, and

2. An OPERABLE flow path capable of recirculating water from the suppression pool through a heat exchanger.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With one suppression pool cooling subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With both suppression pool cooling subsystems inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN[a] within the next 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

M. Suppression Pool Cooling

The suppression pool cooling function of the RHR system shall be demonstrated OPERABLE:

1. At least once per 31 days by verifying that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

2. By verifying that each of the required RHR pumps develops the required recirculation flow through the heat exchanger and the suppression pool when tested pursuant to Specification 4.0.E.

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a Whenever the two required RHR SDC mode subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

3.7 - LIMITING CONDITIONS FOR OPERATION

N. SECONDARY CONTAINMENT INTEGRITY

SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY:

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

ACTION:

1. Without SECONDARY CONTAINMENT INTEGRITY in OPERATIONAL MODES(s) 1, 2 or 3, restore SECONDARY CONTAINMENT INTEGRITY within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. Without SECONDARY CONTAINMENT INTEGRITY in OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

4.7 - SURVEILLANCE REQUIREMENTS

N. SECONDARY CONTAINMENT INTEGRITY

SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

1. Verifying at least once per 24 hours that the pressure within the secondary containment is ≥0.25 inches of vacuum water gauge.

2. Verifying at least once per 31 days that:

   a. At least one door in each secondary containment air lock is closed.

   b. All secondary containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic dampers secured in position.

3. At least once per 18 months by operating one standby gas treatment subsystem at a flow rate ≤4000 cfm for one hour and maintaining ≥0.25 inches of vacuum water gauge in the secondary containment.

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* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.7 - LIMITING CONDITIONS FOR OPERATION

O. Secondary Containment Automatic Isolation Dampers

Each secondary containment ventilation system automatic isolation damper shall be OPERABLE.

APPLICABILITY:

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

ACTION:

With one or more of the secondary containment ventilation system automatic isolation dampers inoperable, maintain at least one isolation damper OPERABLE in each affected penetration that is open and within 8 hours either:

1. Restore the inoperable damper(s) to OPERABLE status, or

2. Isolate each affected penetration by use of at least one deactivated automatic damper secured in the isolation position, or

3. Isolate each affected penetration by use of at least one closed manual valve or blind flange.

Otherwise, in OPERATIONAL MODE(s) 1, 2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Otherwise, in OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE

4.7 - SURVEILLANCE REQUIREMENTS

O. Secondary Containment Automatic Isolation Dampers

Each secondary containment ventilation system automatic isolation damper shall be demonstrated OPERABLE:

1. Prior to returning the damper to service after maintenance, repair, or replacement work is performed on the valve or its associated actuator, control, or power circuit by performance of a cycling test.

2. At least once per 18 months by verifying that on an isolation test signal each automatic isolation damper actuates to its isolation position.

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* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.7 - LIMITING CONDITIONS FOR OPERATION

ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

4.7 - SURVEILLANCE REQUIREMENTS

3.7 - LIMITING CONDITIONS FOR OPERATION

P. Standby Gas Treatment System

Two independent standby gas treatment subsystems shall be OPERABLE, each with an OPERABLE diesel generator power source.

APPLICABILITY:

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

ACTION:

1. With one standby gas treatment subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or:

   a. In OPERATIONAL MODE(s) 1,2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   b. In OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

2. THIS ITEM INTENTIONALLY LEFT BLANK

4.7 - SURVEILLANCE REQUIREMENTS

P. Standby Gas Treatment System

Each standby gas treatment subsystem shall be demonstrated OPERABLE:

1. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters operating.

2. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the subsystem by:

   a. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of <1% and uses the test procedure guidance in Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 4000 cfm ±10%.

   b. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM-D-3803-89, for a methyl iodide penetration of <10%, when tested at 30°C and 70% relative humidity; and

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.7 - LIMITING CONDITIONS FOR OPERATION

3. THIS ITEM INTENTIONALLY LEFT BLANK

4.7 - SURVEILLANCE REQUIREMENTS

c. Verifying a subsystem flow rate of 4000 cfm ±10% during system operation when tested in accordance with ANSI N510-1980.

3. After every # hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM-D-3803-89, for a methyl iodide penetration of <10%, when tested at 30°C and 70% relative humidity.

4. At least once per 18 months by:

a. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is <6 inches water gauge while operating the filter train at a flow rate of 4000 cfm ±10%.

b. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

1) Manual initiation from the control room, and

2) Simulated automatic initiation signal.

c. Verifying that the heaters dissipate 30 ±3 kw when tested in accordance with ANSI N510-1980. This reading shall include the appropriate correction for variations from 480 volts at the bus.

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3.7 - LIMITING CONDITIONS FOR OPERATION

4. With both standby gas treatment subsystems otherwise inoperable in OPERATIONAL MODE(s) 1,2 or 3, restore at least one subsystem to OPERABLE status within one hour, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

5. With both standby gas treatment subsystems inoperable in OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

4.7 - SURVEILLANCE REQUIREMENTS

5. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter bank satisfies the in-place penetration and leakage testing acceptance criteria of <1% in accordance with ANSI N510-1980 while operating the system at a flow rate of 4000 cfm ±10%.

6. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the in-place penetration and leakage testing acceptance criteria of <1% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at a flow rate of 4000 cfm ±10%.

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* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

BASES

3/4.7.A PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10CFR Part 100 during accident conditions.

3/4.7.B Primary Containment Leakage

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 48 psig, P_a. As an added conservatism, the measured overall integrated leakage rate, L_a, is further limited to $\leq 0.75\ L_a$ during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests. Periodic testing of the containment boundary is required to verify the allowable leakage rates are met. Generally, these tests are conducted while shutdown and the leakage rates must be verified as acceptable prior to establishing containment integrity. Type B and C tests may, however, be conducted at power and be found to exceed the limit while in the applicable OPERATIONAL MODE(s). A short time frame, consistent with the allowed outage time for PRIMARY CONTAINMENT INTEGRITY, is provided to restore the leakage to within its limits. If the leakage is tested individually for each valve in a penetration, closing and locking the other containment isolation valve with an acceptable leakage rate restores PRIMARY CONTAINMENT INTEGRITY. Locking the valve with an acceptable leakage rate is required to assure that the leakage rate is not exceeded due to a single failure that could cause the valve to be re-opened.

The main steam line isolation valves are tested at lower pressures, per an approved exemption, but the leakage rate is included in the Type B and C test totals. The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10CFR Part 50 with the exception of approved exemptions. (Ref: Exemption Request Approval, Mr. D. B. Vassallo (NRC) to Mr. D. L. Farrar (CECo) dated June 12, 1984.)

3/4.7.C Primary Containment Air Locks

The limitations on closure and leakage for the primary containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the primary containment leakage rate given in Specifications 3.7.A. and 3.7.B. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation.

3/4.7.D Primary Containment Isolation Valves

The OPERABILITY of the primary containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

The containment is also penetrated by a large number of small diameter instrument lines which contact the primary coolant system. A program for periodic testing and examination of the flow check valves in these lines is performed by blowing down the instrument line during a vessel hydro and observing conditions which verify that the flow check valve is operable, e.g., a distinctive 'click' when the poppet valve seats, or an instrumentation high flow that quickly reduces to a slight trickle.

3/4.7.E Suppression Chamber - Drywell Vacuum Breakers

The function of the suppression chamber to drywell vacuum breakers is to relieve vacuum in the drywell. These internal vacuum breakers allow air and steam flow from the suppression chamber to the drywell when the drywell is at a negative pressure with respect to the suppression chamber. Each vacuum breaker is a self-actuating valve, similar to a check valve.

The safety analysis assumes that the internal vacuum breakers are closed initially and are fully open at a differential pressure of 0.5 psid. Additionally, three of these internal vacuum breakers are assumed to fail in a closed position. The results of the analyses show that the design pressure is not exceeded even under the worst case accident scenario.

The vacuum breakers between the suppression chamber and the drywell must not be inoperable in the open position since this would allow bypassing of the suppression pool in case of an accident. There is a sufficient number of valves so that operation may continue for a limited time with up to three vacuum breakers inoperable in the closed position.

Each suppression chamber to drywell vacuum breaker is fitted with a redundant pair of position switches which provide signals of disk position to panel mounted indicators and annunciate an alarm in the control room if the disk is open more than allowable. The alarm systems meet the intent of IEEE-279 standards.

3/4.7.F Reactor Building -Suppression Chamber Vacuum Breakers

The function of the reactor building to suppression chamber vacuum breakers is to relieve vacuum when the suppression chamber atmosphere depressurizes below reactor building pressure. If the drywell depressurizes below reactor building pressure, the negative differential pressure is

mitigated by flow through the reactor building to suppression chamber vacuum breakers and through the suppression chamber to drywell vacuum breakers. The reactor building to suppression chamber vacuum breakers include both an air operated valve and a check valve in each line. However, position indication is only provided on the air operated valve. These lines and valves are sized on the basis of the air flow from the secondary containment that is required to mitigate the depressurization transient and limit the maximum negative drywell pressure to within design limits. The maximum depressurization rate is a function of the drywell spray flow rate and temperature and the assumed initial conditions of the drywell atmosphere. The safety analyses assume the external vacuum breakers to be closed initially and to be fully open at 1.0 psid. Both vacuum breakers are periodically demonstrated to open at the required pressure differential. For the air operated vacuum breaker, this demonstration is essentially a CHANNEL CALIBRATION of the logic system. Additionally, of the two reactor building to suppression chamber vacuum breaker lines, one is assumed to fail in a closed position to satisfy the single active failure criterion.

3/4.7.G Drywell Internal Pressure

The limitations on drywell internal pressure ensure that the containment peak pressure does not exceed the design pressure during the Design Basis Accident (DBA). The upper limit for initial positive containment pressure will limit the total post accident design basis pressure to approximately 48 psig which is less than the design pressure and is consistent with the safety analysis. The maximum pressure, and the minimum pressure above 15% RATED THERMAL POWER, is also based on assumptions for post-accident hydrodynamic loading analysis. A short period is allowed to conduct testing, e.g. HPCI, vacuum breaker and relief valve testing, which temporarily reduces the drywell pressure below this minimum.

3/4.7.H Drywell-Suppression Chamber Differential Pressure

The toroidal-shaped suppression chamber, which contains the suppression pool is connected to the drywell by eight main vent pipes. The main vent pipes exhaust into a vent header, from which downcomer pipes extend into the suppression pool. During a loss-of-coolant accident (LOCA), the increasing drywell pressure will force the water leg in the downcomer pipes into the suppression pool at substantial velocities as the blowdown phase of the event begins. The length of the water leg has a significant effect on the resultant primary containment pressures and loads.

The purpose of maintaining the drywell at a slightly higher pressure with respect to the suppression chamber is to minimize the drywell pressure increase necessary to clear the downcomer pipes to commence condensation of steam in the suppression pool and to minimize the mass of the accelerated water leg. This reduces the hydrodynamic loads on the torus during the LOCA blowdown. Initial drywell-to-suppression-chamber differential pressure affects both the dynamic pool loads on the suppression chamber and the peak drywell pressure during downcomer pipe clearing during a Design Basis Accident. Drywell-to-suppression-chamber differential pressure must be maintained within the specified limits so that the safety analysis remains valid. However, a short period is allowed to conduct testing, e.g. HPCI, vacuum breaker and relief valve testing,

which temporarily increases the suppression chamber pressure and reduces the differential pressure. Only one direct suppression chamber to drywell differential pressure instrumentation CHANNEL is provided. However, any pair of the redundant drywell and suppression chamber pressure instrumentation CHANNEL(s) are sufficient to determine the differential pressure.

3/4.7.I Primary Containment Nitrogen System

The nitrogen system functions to maintain oxygen concentrations within the primary containment at or below the explosive levels. To ensure that a combustible gas mixture does not occur, oxygen concentration is kept below 4.0 volume percent. The system operates in conjunction with emergency operating procedures that are used to reduce primary containment pressure periodically during system operation. This combination results in a feed-and-bleed approach to maintaining hydrogen and/or oxygen concentrations below combustible levels. Sufficient liquid nitrogen is maintained to provide approximately a seven day supply to allow for establishing an additional nitrogen supply following a LOCA.

3/4.7.J Primary Containment Oxygen Concentration

All nuclear reactors must be designed to withstand events that generate hydrogen either due to the zirconium metal-water reaction in the core or due to radiolysis. The primary method to control hydrogen is to inert the primary containment. With the primary containment inerted, that is, oxygen concentration less than 4.0 volume percent, a combustible mixture cannot be present in the primary containment for any hydrogen concentration. The Design Basis Accident (DBA) loss-of-coolant accident (LOCA) analysis assumes that the primary containment is inerted when the DBA occurs. Thus, the hydrogen assumed to be released to the primary containment as a result of a metal-water reaction in the reactor core will not produce combustible gas mixtures in the primary containment.

The primary containment oxygen concentration must be within the specified limit when primary containment is inerted, except as allowed by the relaxations during startup and shutdown. The primary containment must be inert in OPERATIONAL MODE 1, since this is the condition with the highest probability of an event that could produce hydrogen. Inerting the primary containment is an operational problem because it prevents containment access without an appropriate breathing apparatus. Therefore, the primary containment is inerted as late as possible in the plant startup and de-inerted as soon as possible in the plant shutdown. As long as reactor power is below 15% of RATED THERMAL POWER, the potential for an event that generates significant hydrogen is low and the primary containment does not need to be inert. Furthermore, the probability of an event that generates hydrogen occurring within the first 24 hours of a reactor startup or within the last 24 hours before a shutdown is low enough that these windows, when the primary containment is not inerted, are also justified. The 24 hour time frame is a reasonable amount of time to allow plant personnel to perform inerting or de-inerting.

3/4.7.K Suppression Chamber

The specifications of this section ensure that the primary containment pressure will not exceed the design pressure during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from ~1000 psig. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss of coolant accident, the pressure of the liquid and gas must not exceed the suppression chamber maximum pressure. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

An allowable bypass area between the primary containment and the drywell and suppression chamber is identified based on analysis considering primary system break area, suppression chamber effectiveness, and containment design pressure. Analyses show that the maximum allowable bypass area is equivalent to all vacuum breakers open the equivalent of 1/16 inch at all points along the seal surface of the disk.

Using the minimum or maximum water levels given in this specification (as measured from the bottom of the suppression chamber), primary containment maximum pressure following a design basis accident is approximately 48 psig, which is below the design pressure. The maximum water level results in a downcomer submergence of 4 feet and the minimum level results in a submergence approximately 4 inches less. If it becomes necessary to make the suppression chamber inoperable, it is done in accordance with the requirements in Specification 3.5.C.

Because of the large volume and thermal capacity of the suppression pool, the level and temperature normally change very slowly and monitoring these parameters daily is sufficient to establish any trend. By requiring the suppression pool temperature to be more frequently monitored during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

Under full power operating conditions, blowdown from an initial suppression chamber water temperature of 95°F results in a water temperature of approximately 145°F immediately following blowdown which is low enough to provide complete condensation via T-quencher devices. At this temperature and atmospheric pressure, the available net positive suction head exceeds that required by the emergency core cooling system pumps, thus there is no dependency on containment overpressure during the accident injection phase.

Experimental data indicates that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained sufficiently low during any period of safety relief valve operation for T-quencher devices. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings. In addition to the limits on temperature of the suppression chamber pool water, operating procedures define the action to be taken in the event a safety or relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, (3) initiate reactor shutdown, and (4) if other safety or relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety or relief valve to assure mixing and uniformity of energy insertion to the pool.

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed which demonstrated a factor of safety of at least two for the weakest element in the suppression chamber support system and attached piping. The maintenance of a drywell-suppression chamber differential pressure and a suppression chamber water level corresponding to a downcomer submergence range of 3.67 to 4.00 feet will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

3/4.7.L Suppression Chamber and Drywell Spray

Following a Design Basis Accident (DBA), the suppression chamber spray function of the containment cooling mode of the residual heat removal (RHR) system removes heat from the suppression chamber air space and condenses steam. The suppression chamber is designed to absorb the sudden input of heat from the primary system from a DBA or a rapid depressurization of the reactor pressure vessel through safety or relief valves. There is one 100% capacity containment spray header inside the suppression chamber. Periodic operation of the suppression chamber and drywell sprays may also be used following a DBA to assist the natural convection and diffusion mixing of hydrogen and oxygen when other ECCS requirements are met and oxygen concentration exceeds 4%. Since the spray system is a function of the RHR system, the loops will not be aligned for the spray function during normal operation, but all components required to operate for proper alignment must be OPERABLE.

3/4.7.M Suppression Pool Cooling

Following an accident, the suppression pool cooling function of the RHR system removes heat that the suppression pool absorbs from the primary system and, in the long term, continues to absorb residual heat generated by fuel in the reactor core. Each of the suppression pool cooling loops consists of a pump and heat exchanger. Following a loss of coolant accident (LOCA), the plant operators can realign the valves in these two loops to draw water from the suppression pool, pump it through the shell side of the exchangers, and discharge it back to the suppression pool via the full flow test lines. At the same time, residual heat removal service water (RHRSW) is pumped through the tube side of the exchangers to exchange heat to the external heat sink.

3/4.7.N SECONDARY CONTAINMENT INTEGRITY

The function of the secondary containment is to isolate and contain fission products that escape from primary containment following a Design Basis Accident (DBA), to confine the postulated release of radioactive material within the requirements of 10CFR Part 100, and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside of primary containment. The reactor building and associated structures provide secondary containment during normal operation when the drywell is sealed and in service. At other times the drywell may be open and, when required, secondary containment integrity is specified. There are two principal accidents for which credit is taken for secondary containment OPERABILITY. These are a LOCA and fuel-handling accident inside secondary containment. The secondary containment performs no active function in response to each of these limiting events; however, its leak tightness is required to limit offsite radiation doses to below those required by 10CFR Part 100. Maintaining secondary containment OPERABLE ensures that the release of radioactive materials from the primary containment is restricted to those leakage paths and associated leakage rates assumed in the accident analysis and that fission products entrapped within the secondary containment structure will be treated prior to discharge to the environment. Establishing and maintaining a vacuum in the reactor building with the standby gas treatment system during testing, along with the surveillance of the doors, hatches, dampers and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment. This surveillance is normally conducted during periods of calm winds (<5 mph), but may be conducted under higher wind conditions with appropriate application of correction factors.

3/4.7.O Secondary Containment Automatic Isolation Dampers

The function of the secondary containment ventilation system automatic isolation dampers, in combination with other accident-mitigation systems, is to limit fission-product release during and following postulated Design Basis Accidents (DBA) such that offsite radiation exposures are maintained within the requirements of 10CFR Part 100. Secondary containment isolation ensures that fission products that escape from primary containment following a DBA, or which are released during certain operations when primary containment is not required, or take place outside primary containment, are maintained within applicable limits. The OPERABILITY requirements for the secondary containment ventilation system isolation dampers help ensure that adequate secondary containment leak tightness is maintained during and after an accident by minimizing potential paths to the environment.

3/4.7.P Standby Gas Treatment System

The standby gas treatment system (SBGT) is required to ensure that radioactive materials that leak from the primary containment into the secondary containment following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment. This system reduces the

potential releases of radioactive material, principally iodine, to within values specified in 10CFR Part 100.

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the main chimney during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environment. One standby gas treatment fan is designed to automatically start upon secondary containment isolation and to maintain the reactor building pressure to approximately a negative ¼ inch water gauge pressure; all leakage should be in-leakage. Should the fan fail to start, the redundant alternate fan and filter subsystem is designed to start automatically.

The OPERABILITY of the standby gas treatment system reduces the potential release of radioactive material, principally iodine, following a design basis accident. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses. Periodic operation of the system with the heaters is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.

Since the standby gas treatment subsystems are shared by both units, one subsystem is powered by the unit diesel generator power source of each unit. This unique arrangement requires that special allowed out-of-service times be provided for the combinations of subsystem and diesel generator power source inoperability that may occur. For example, if conducting the alternate offsite power source cross-tie surveillance were to require the inoperability of both unit diesel generator power sources, neither of the standby gas treatment subsystems would have an OPERABLE diesel generator power source and the appropriate ACTION would have to be entered.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 143 TO FACILITY OPERATING LICENSE NO. DPR-19,

AMENDMENT NO. 137 TO FACILITY OPERATING LICENSE NO. DPR-25,

AMENDMENT NO. 165 TO FACILITY OPERATING LICENSE NO. DPR-29,

AND AMENDMENT NO. 161 TO FACILITY OPERATING LICENSE NO. DPR-30

COMMONWEALTH EDISON COMPANY

AND

MIDAMERICAN ENERGY COMPANY

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

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

1.0 INTRODUCTION

By letter dated September 17, 1993, as supplemented by letter dated July 20, 1995, Commonwealth Edison Company (ComEd, the licensee) submitted an amendment requesting to upgrade sections of the Dresden Nuclear Power Station, Units 2 and 3, and the Quad Cities Nuclear Power Station, Units 1 and 2, Technical Specifications (TS). The changes have been requested as part of their Technical Specification Upgrade Program (TSUP).

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

The licensee made the decision to initiate a TSUP for both Dresden and Quad Cities. The licensee evaluated the current TS for both stations against the Standard Technical Specifications (STS), contained in NUREG-0123, "Standard Technical Specifications General Electric Plants BWR/4, 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.

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

The application dated September 17, 1993, as supplemented July 20, 1995, proposed to upgrade only those sections of the TS to be included in TSUP Section 3/4.7 (Containment Systems) of the Dresden and Quad Cities TS.

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

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

2.0 EVALUATION

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

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

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 reflect industry lessons learned in the 1980's and early 1990's.

Deviations between the proposed specifications, the STS, and the current TS were reviewed by the staff to determine if they were due to plant specific features or if they posed a technical deviation from the STS guidelines. Plant specific data, values, parameters, and equipment specific operational requirements contained in the current TS for Dresden and Quad Cities were retained by the licensee in the upgraded TS. Portions of the proposed TS may involve issues which have not been resolved and therefore not approved by the staff. These issues will remain open items and will be addressed by the licensee in a clean-up amendment request. Upon receipt and review of this submittal, the staff will issue a final amendment which addresses each of the open items.

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 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 requires applicants for nuclear power plant operating licenses to state TSs 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 following sections provide the staff's evaluations of the specific proposed TS changes.

3.0 EVALUATION OF TSUP PROPOSED TS SECTION 3/4.7

The staff reviewed the proposed TS against the current TS and the STS guidance. The deviations between the proposed TS and the current TS and STS are evaluated below.

3.1 3/4.7.A. Primary Containment Integrity

Proposed TS Section 3/4.7.A incorporates the requirements of STS Section 3/4.6.1.1 and current TS Section 3/4.7.A.2 for both stations. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR.

3.1.1 Applicability

The proposed applicability deviates from the current TS regarding the exception for performing low power testing. The current TS specify that the power level should not exceed 5 MWt (0.2 percent power) during this testing. The proposed TS specify one percent of rated thermal power (25 MWt). One percent of rated thermal power is a more measurable quantity as compared to 5 MWt and can be more accurately differentiated from core decay heat. The proposed requirement has been shown based on industry precedent and experience to provide an adequate level of protection in assuring that primary containment integrity is maintained during low power physics tests. The proposed TS references proposed TS section 3.12.A which provides enhanced guidance to site operations personnel by including specific actions, applicability and surveillances not included in the current TS to ensure potential degraded conditions associated with primary containment integrity associated with low power physics tests are appropriately addressed. Therefore, the proposed applicability statement is acceptable.

3.1.2 LCO

The current TS requirements concerning primary containment integrity in Section 3/4.7.A.2 of the Dresden and Quad Cities TS have been incorporated into the proposed TS LCO. The staff finds the proposed LCO has been formatted in accordance with the STS guidelines. Therefore, the staff finds the proposed LCO for proposed TS Section 3/4.7.A to be acceptable.

3.1.3 Required Actions

The current TS do not contain specific action statements. Therefore, the plant defaults to TS 3.0.A which requires the plant be brought to cold

shutdown within 24 hours. The proposed TS relaxes this requirement by providing a 1 hour allowed outage time (AOT) before bringing the plant to hot shutdown in the following 12 hours. The AOT is provided in order to allow the plant some period of time to restore a potentially degraded condition and does not significantly affect the level of safety of the current TS. The requirement to bring the plant to hot shutdown provides an equivalent level of safety as bringing the plant to cold shutdown because it brings the plant to a condition in which the TS no longer applies. In addition, the proposed TS ensures that operator action is initiated in a more expeditious time frame. The proposed TS is not a change in the current design requirements and is acceptable.

3.1.4 Surveillance Requirements

Proposed TS 4.7.A.1 through 4.7.A.4 are new requirements adopted from STS Section 4.6.1.1. Proposed TS 4.7.A.1 requires leak testing for penetrations opened following a Type A or B test. The test conditions and test acceptance criteria are consistent with those in proposed TS 3.7.B, "Primary Containment Leakage". Proposed TS 4.7.A.2 requires monthly verification that all primary containment penetrations not capable of being closed by operable containment automatic isolation valves and which are required to be closed during accident conditions, are closed. These are new requirements which enhance the current TS and are acceptable. Proposed TS 4.7.A.3 and 4.7.A.4 require verification of containment air lock operability and suppression chamber operability and refer to the associated TS for these components (proposed TS 3.7.C and 3.7.K). These TS are discussed in the sections below.

3.2 3/4.7.B, Primary Containment Leakage

Proposed TS Section 3/4.7.B incorporates the requirements of STS Section 3/4.6.1.2 and current TS Section 3/4.7.A.2 for both stations. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR.

3.2.1 Applicability

The current applicability requirements have been incorporated into the proposed TS. The staff finds the proposed applicability statement has incorporated all current TS requirements and has been formatted in accordance with the STS guidelines. Therefore, the staff finds the applicability for proposed TS Section 3/4.7.B to be acceptable.

3.2.2 LCO

Current TS 3.7.A.2.a allows reduced pressure tests (Pt) as an alternative measure for satisfying the LCO requirements for Primary Containment Leakage requirements. This is an alternative to full pressure tests. The proposed TS conservatively eliminates this option. The elimination of the reduced pressure tests eliminates uncertainties associated with the correlation to

full pressure test data. The proposed TS is more conservative and is therefore acceptable.

Current Dresden TS 3.7.A.2.a(5) provides the definition of total measured leakage rates. This definition is encompassed within 10 CFR Part 50, Appendix J, and its inclusion in the TS is unnecessary. Therefore, the deletion of the definition in the proposed TS is acceptable. Current Quad Cities TS do not contain this specification.

Current Dresden TS 3.7.A.2.b(1) and Quad Cities TS 3.7.A.2.b require that overall integrated leakage rate for Type A tests be $\leq 0.75\ L_a$. The proposed TS move this requirement from an LCO to a surveillance requirement. Proposed TS 4.7.B.2 requires that Type A tests be performed and meet the $\leq 0.75\ L_a$ criteria. The proposed TS LCO doesn't contain a separate requirement for Type A test leakage. The proposed LCO limit of 1.0 L_a is the applicable limit. Current TS 3.7.A.2.b(1) and 3.7.A.2.b are intended to ensure that a successful Type A test has been conducted prior to resuming power operations so that a margin is provided to ensure maintenance of containment integrity during plant operations. This limit is applicable only during performance of periodic Type A tests to account for anticipated degradation due to plant operations subsequent to each leakage test. The purpose of the current TS is fulfilled by the proposed TS. Therefore, the proposed TS is acceptable.

STS 3.6.1.2.d limits the combined leakage rate for all containment isolation valves in hydrostatically tested lines which penetrate primary containment. Because the current primary containment design basis for Dresden and Quad Cities does not require hydrostatic testing of primary containment penetrations, the proposed TS do not include this requirement.

3.2.3 Required Actions

The current TS contain an exception which excludes main steam isolation valve (MSIV) leakage from the total combined leakage limit in the action statement. The proposed TS deletes this exception since MSIV leakage will be added to total leakage for the purposes of meeting the action statement criteria. The proposed limit is equivalent to the current limit of 0.60 L_a. Therefore, by including the MSIV leakage in the action statement total, the proposed TS is more conservative than current TS and is acceptable.

The proposed TS include a 1 hour AOT to restore the leakage rate to within limits. There is no such guidance in the current Quad Cities TS. In addition, there are no action requirements in the current Dresden TS and the plant would defer to TS 3.0.A. The proposed TS, therefore, is a relaxation of current requirements. The proposed AOT is consistent with the proposed primary containment integrity specifications (proposed TS 3.7.A). The one hour period is an acceptable amount of time to restore leakage before initiating a shutdown. The relaxation of current requirements maintains the design requirements and is acceptable.

3.2.4 Surveillance Requirements

The proposed TS deletes the current Quad Cities requirement to use the methods and provisions of the American National Standards Institute (ANSI) N45.4 for leak rate testing. The specific ANSI requirements for primary containment leakage rate testing are outlined in 10 CFR Part 50, Appendix J. It is unnecessary to repeat this reference in the TS. The actual testing requirement is not affected by this change and the proposed TS is consistent with current Dresden TS. Therefore, the proposed SR is acceptable.

The current TS allow the use of either injection of gas into containment or bleeding gas from containment for the Type A test verification. The proposed TS deletes the option of gas injection since this option is not used at either Dresden or Quad Cities. The proposed TS does not relax any current requirements and is acceptable.

Current Quad Cities TS 4.7.A.2.e discusses the methodology for leakage rate data collection and has been deleted in the proposed TS. Current Dresden TS do not contain this specification. The specific methodology used to satisfy SRs is inappropriate for inclusion in the TS. These details are more appropriate for inclusion in station procedures. The staff has determined that the requirements for leakage rate data collection 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 Dresden TS 4.7.A.2.f requires continuous leak rate monitoring when the containment is inerted by reviewing the inerting system make-up requirements. Current Quad Cities TS do not contain this specification. The proposed TS have not retained this requirement. An acceptable primary containment leakage rate is ensured in the proposed TS by the reference to Appendix J of 10 CFR Part 50. By following the required testing of Appendix J, the safety margin associated with containment leakage rates is maintained. These requirements include both Type A overall integrated containment leakage rate tests and Type B and C tests for containment penetrations. The staff has determined that the requirements for continuous leak rate 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 Part 50, Appendix J, to assure continued protection of public health and safety. Therefore, this change is acceptable.

The guidelines of STS 4.6.1.2.d.3 through 4.6.1.2.d.6 and 4.6.1.2.e through 4.6.1.2.k have not been incorporated into the proposed TS due to plant design limitations. Dresden and Quad Cities have no penetrations which are continuously monitored or hydrostatically tested, nor do purge valves use resilient seals. Current TS requirements are not relaxed by not incorporating the aforementioned STS requirements and, therefore, the proposed TS do not affect existing plant safety margins and are acceptable.

3.3 3/4.7.C, Primary Containment Airlocks

Proposed TS Section 3/4.7.C incorporates the requirements of STS Section 3/4.6.1.3 and current TS Section 3/4.7.A.8 for Dresden and 3/4.7.A.7 for Quad Cities. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR.

3.3.1 LCO and Applicability

The proposed LCO is consistent with current requirements and is, therefore, acceptable. The proposed applicability is Modes 1, 2, and 3 as modified by Special Test Exception 3.12.A. The current TS do not provide explicit applicability requirements. However, the current TS require that the plant be brought to cold shutdown (Mode 4) if the LCO can not be met. Thus the current LCO is applicable in Modes 1, 2, and 3. The proposed applicability is consistent with the current TS and is acceptable.

3.3.2 Required Actions

Proposed Action 1 requires that with one air lock door inoperable, the door must be restored or locked closed within 24 hours and verified locked closed every 31 days. Proposed Action 2 requires that with the air lock interlock mechanism inoperable, it must be restored within 24 hours or one door must be locked closed and verified closed every 31 days. Action 3 requires that with the air lock inoperable except as a result of an inoperable door or interlock mechanism, the air lock must be restored within 24 hours or the reactor must be in hot shutdown in 12 hours and cold shutdown in the following 24 hours. These actions are consistent with current TS requirements and are acceptable.

3.3.3 Surveillance Requirements

The proposed surveillance requirements require an air lock leakage test within 72 hours of opening, at least once per 6 months, and prior to establishing primary containment integrity following air lock opening (including verification that only one door in each airlock can be opened at a time). These requirements are consistent with current TS and are acceptable.

3.4 3/4.7.D, Primary Containment Isolation Valves

Proposed TS Section 3/4.7.D incorporates the requirements of STS Section 3/4.6.3 and current TS Section 3/4.7.D for both stations. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR.

3.4.1 LCO

The current Quad Cities TS contain a table which lists all applicable isolation valves. The proposed TS have deleted this table and the reference to it consistent with the guidance of GL 91-08, "Component Lists Relocation." The listing of the containment isolation valves in the TS is redundant to the

listing in an owner controlled document which was created to comply with the provisions of GL 91-08. The inclusion of this list in the owner controlled document maintains the equivalent level of protection for the plants. The current TS also specify the reference leg backfill check valves as valves required to be operable per this TS. These valves are addressed as primary containment isolation valves in the owner controlled document. It is unnecessary to specifically list these valves in the TS as their inclusion within the component listings created by GL 91-08 maintains the equivalent level of protection for the plants. These changes are in conformance with GL 91-08 and are therefore acceptable. The current Dresden TS do not contain a listing of valves and, therefore, there is no change in the proposed Dresden TS.

The current TS require that the temperature of the main steamline air pilot valves be less than 170 degrees Fahrenheit. This requirement and the associated action statements and SRs have not been retained in the proposed TS. These requirements were added to the current TS to control main steam isolation valve (MSIV) closure failures. The failures are based on the moist, dirty, film buildup experienced in the internals of the pilot assembly which contained internal metal-to-metal contact points. The heat in the containment contributed to the hardening of the film within the pilot. Subsequent to the addition of these requirements in current TS, instrument air quality was improved using dryers. The pilot assemblies were replaced with pilot assemblies with elastomer internals, not susceptible to film buildup. The current provisions are no longer needed since the quality of instrument air has improved and the pilot assemblies associated with the failed MSIV closures have been replaced with assemblies that are not susceptible to pilot valve internal sticking. The current Dresden TS contain a requirement for a bi-weekly partial closure test of the MSIVs. This requirement has been deleted since these tests represent an unnecessary risk to plant availability for a safety benefit that has been superseded by plant modification. The proposed TS are consistent with industry practice and removal of the antiquated provision which required pilot temperature monitoring and bi-weekly valve exercising is both appropriate and acceptable. The proposed periodicity for MSIVs and other containment isolation valves is adequately controlled per the provisions of 4.0.E (which requires inservice testing). The required actions and SRs of proposed TS Section 3/4.7.D have been shown based on industry experience to provide an adequate level of monitoring containment isolation valves. Therefore, the deletion of this requirement is acceptable.

3.4.2 Applicability

The proposed applicability is consistent with the current TS applicability (Modes 1 and 2) and has been conservatively expanded to include Mode 3. The proposed TS provides an equivalent or greater level of protection for the primary containment isolation valves as compared to the current TS and is, therefore, acceptable.

3.4.3 Required Actions

The proposed TS relaxes the current requirements when one or more isolation valves is inoperable by including a four hour AOT to restore the valves to operable status or isolate and deactivate the affected valves. The current TS do not provide specific guidance or explicit time constraints to isolate the affected penetration. The proposed TS requires that, during the four hour AOT, one valve in each affected penetration remains operable. Therefore, the proposed TS provides an adequate level of protection to ensure that the primary containment penetrations are protected. The proposed TS is consistent with STS guidelines.

If the remaining isolation valve in the affected penetration can not be isolated, the current TS require that the plant be in cold shutdown within 24 hours. The proposed TS requires the plant to be hot shutdown in 12 hours and cold shutdown in the following 24 hours under these circumstances. Therefore, the proposed actions allow 16 hours (including the 4 hour AOT) prior to taking the plant out of operating conditions as compared to the current TS allowance of 24 hours to place the plant in cold shutdown conditions. Placing the reactor in hot shutdown puts it in a condition for which the TS is not currently applicable and therefore, in a safe condition. In accordance with the proposed applicability, the proposed TS require that cold shutdown be achieved within the following 24 hours. The proposed requirements have been shown, based on industry experience, to provide an adequate level of protection for ensuring appropriate actions are taken to disposition concerns associated with inoperable primary containment isolation valves. Therefore, the proposed TS is acceptable.

3.4.4 Surveillance Requirements

Current TS 4.7.D.1.c(1) specifies quarterly testing of all normally open power operated isolation valves (except for MSIVs). Proposed TS 4.7.D.3 changes the frequency from quarterly to "pursuant to Specification 4.0.E." TS 4.0.E states that SRs for inservice testing be performed at the frequency specified by Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code except where relief has been granted by the Commission. The power operated isolation valves are currently included in the IST program and the current IST frequency is quarterly. Therefore, the frequency does not change. Proposed TS 4.7.D.3 does not change any requirements but rather, indicates that the test frequency will be in accordance with the IST program. Revisions to the IST program are controlled by the requirements of 10 CFR 50.55a. The staff finds the proposed TS acceptable.

Current TS 4.7.D.1.c(2) regarding MSIV closure times has been relocated to proposed TS 4.6.M which is discussed in a separate safety evaluation. This change is administrative and is acceptable.

Current Dresden TS 4.7.D.1.d, 4.7.D.3, and 4.7.D.4 have not been retained in the proposed TS. These are surveillances of the main steamline air pilot valves. The associated LCO has been deleted as discussed in Section 3.4.1 of

this SE. Therefore, the SRs are no longer required. These changes are acceptable.

Current TS 4.7.D.2 requires the daily recording of valve position for valves in each line with an inoperable valve. This requirement has been deleted in the proposed TS. Proposed Action 1.b requires the isolation of affected penetrations by the use of one deactivated automatic valve. This compensates for manual verification of valve position. Therefore, the proposed TS provides an equivalent level of protection for minimizing the potential risk as compared to current requirements and is acceptable.

The proposed TS adds a new requirement based on STS guidelines. Proposed TS 4.7.D.1 requires demonstration of operability of the primary containment isolation valves (PCIV) following maintenance. This SR is an enhancement to the current TS and is acceptable.

Proposed TS 4.7.D.5 is an enhancement of the current TS. The proposed TS provides explicit guidance for testing of the traversing in-core probe (TIP) system explosive isolation valves. In the current TS, the TIP explosive isolation valves are included in the requirement to verify automatic isolation of all containment isolation valves once per operating cycle (approximately 18 months). The proposed TS requires verification every 31 days of the continuity of the explosive charge and initiation of one explosive squib from each valve every 18 months, with each explosive squib tested at least once per 36 months. The proposed TS is consistent with industry practices and provides explicit guidance appropriate to this type of valve. Therefore, the proposed TS is an enhancement of current TS and is acceptable.

3.5 3/4.7.E, Suppression Chamber - Drywell Vacuum Breakers

Proposed TS Section 3/4.7.E incorporates the requirements of STS Section 3/4.6.4.1 and current TS Section 3/4.7.A.4 for both stations. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR.

3.5.1 LCO

The proposed TS is consistent with the current requirements that nine of the twelve suppression chamber - drywell vacuum breakers be operable and all twelve be closed. This deviates from STS guidelines which require all vacuum breakers to be operable. This is due to the design of Dresden and Quad Cities which only requires nine vacuum breakers to provide sufficient pressure suppression protection. Because the proposed TS is consistent with current requirements it is acceptable.

3.5.2 Applicability

The proposed applicability of Modes 1, 2, and 3 encompasses the current TS which require the suppression chamber - drywell vacuum breakers whenever primary containment is required. Current TS require primary containment

integrity be maintained whenever the reactor is critical or reactor water temperature is greater than 212 degrees Fahrenheit (Modes 1, 2, and 3). The proposed TS is consistent with current requirements and is, therefore, acceptable.

3.5.3 Required Actions

Proposed Action 1 requires that with one of the required vacuum breakers inoperable for opening, but known to be closed, nine vacuum breakers must be restored to operable status within 72 hours or be in at least hot shutdown in 12 hours and cold shutdown in 24 hours. The current TS do not contain a specific action statement for this condition and, therefore, the current TS default to TS 3.0.A which requires cold shutdown in 24 hours. Therefore, the proposed TS is a relaxation of current requirements. However, the proposed action requirements only apply if the vacuum breaker is verified to be closed. The vacuum breakers between the suppression chamber and the drywell must not be inoperable in the open position since this would allow bypassing of the suppression pool in case of an accident. The 72 hour AOT provides a reasonable period of time to restore the full function of the vacuum breaker while not affecting the path to the suppression pool in case of an accident. Therefore, the proposed relaxation from current TS does not significantly reduce existing plant safety margins. If the breakers are open, a 4 hour AOT is allowed after which time the plant is required to be in hot shutdown within 12 hours and cold shutdown within the following 24 hours. The requirements to restore an open vacuum breaker to closed position within 4 hours provides enhanced requirements to operators that appropriately address this situation. Therefore, existing safety margins are not affected. Therefore, the proposed action is acceptable.

Proposed Action 2 is a new requirement not in current TS. This requirement states that with one vacuum breaker open, restore the open vacuum breaker to the closed position within 4 hours or be in at least hot shutdown in 12 hours and cold shutdown in the following 24 hours. Because the current TS do not address this situation, the current TS would require the plant to be in hot shutdown in 12 hours and cold shutdown in the following 24 hours. The proposed TS deviates from the STS guidance which specifies a 2-hour AOT to verify that the second vacuum breaker in series is closed. The difference in action statements reflects the design of Dresden and Quad Cities which features single vacuum breakers rather than a pair in series. When a single vacuum breaker is determined to be open, the proposed TS action is to close the vacuum breaker. Four hours is considered the minimum time necessary to safely plan and complete the manual cycling necessary to close the vacuum breaker which is located in a high radiation area. The proposed TS allows an appropriate amount of time to complete repairs and possibly avoid a shutdown. If the vacuum breaker cannot be repaired within the 4 hours, sufficient time (12 hours) remains to perform a controlled shutdown. The proposed TS accurately reflects the design of Dresden and Quad Cities and does not result in a decrease in safety. Therefore, the proposed TS is acceptable.

Proposed Action 3 discusses the requirement when one position indicator of any operable vacuum breaker is inoperable. The proposed action for Dresden

deviates from the current requirements. The current TS allow one position alarm circuit in an operable vacuum breaker to be inoperable for 15 days provided that each vacuum breaker is physically verified to be closed immediately and daily thereafter. The proposed TS allows one position indicator to be inoperable for an indefinite period of time provided that the vacuum breaker is verified to be closed at least once per 24 hours. If the manual verification is not performed, after 14 days the breaker is considered inoperable. The 14 day period is consistent with STS guidelines which only require verification of breaker position every 15 days. The indefinite period of operation, if daily verification is made, is based on the fact that the position indicator performs an indication-only function and is not assumed in any event for dependence on operator action. An indefinite period of operation is acceptable provided an alternate means of verifying that the vacuum breaker is closed is performed every 24 hours. The proposed TS reflects the single vacuum breaker design in each line with dual position indication. The proposed TS deviate from STS guidelines which recommend a differential pressure test be conducted to verify that the breaker is closed every 15 days. The proposed TS also verify that the breaker is closed but at more frequent intervals. Therefore, the proposed method provides an equivalent or greater level of safety and is acceptable. Based on the above discussion, the proposed TS does not significantly reduce existing plant safety margins and is acceptable.

The current and proposed Quad Cities TS allow indefinite operation with one position indicator inoperable provided that verification is made within 24 hours and every 15 days that the vacuum breaker is closed. Because the Quad Cities design precludes visually verifying closure at operating conditions, the current and proposed action for Quad Cities requires that a test be conducted which demonstrates that the differential pressure is maintained at greater than or equal to 0.5 psi for 1 hour without makeup. The proposed TS is consistent with current requirements and is, therefore, acceptable.

3.5.4 Surveillance Requirements

Proposed TS 4.7.E.2.a is a new requirement not in current TS. The proposed SR requires cycling of each vacuum breaker at least once per 31 days and within 12 hours after any discharge of steam from one or more main steam relief valves (MSRVs). This deviates from STS guidelines which requires the surveillance within 2 hours after discharge from an MSRV. A 12 hour time period provides a reasonable time limit to perform this surveillance and is consistent with the guidance provided in GL 93-05. The proposed TS is an enhancement of current requirements and is, therefore, acceptable.

Current Quad Cities TS 4.7.A.4.a(2) which requires a differential pressure decay test, has not been retained in proposed TS 4.7.E. The differential pressure decay test is used as a contingency requirement to satisfy the action of proposed TS 3.7.E, Action 3 if normal position indication for the vacuum breakers is inoperable. It is unnecessary and redundant to routinely perform the differential pressure decay test when normal vacuum breaker position indication is operable. The proposed requirements are consistent with industry standards which have been shown to provide an adequate level of protection for ensuring suppression chamber - drywell vacuum breaker position

indication is maintained. Because the differential pressure decay test is used as contingency in lieu of normal position indication for ensuring adequate suppression chamber - drywell differential pressure requirements exist, the deletion of this requirement does not reduce existing plant safety margins and is acceptable.

Current TS 4.7.A.4.b(3) which requires inspection of 25 percent of the vacuum breakers at every refueling outage has not been retained in proposed TS 4.7.E. The current requirement has no specific inspection criteria associated with it. This requirement is currently used to inspect breakers which are found to be deficient during the other surveillances conducted during refueling outages. The current inspection requirement is redundant to the surveillances required by current TS 4.7.A.4.b(1) and (2) (proposed TS 4.7.E.2.c.1 and c.2). If vacuum breakers are discovered to be deficient during these surveillances, additional inspection would necessarily be required to determine the cause. Additional inspections of operable vacuum breakers do not provide any new information regarding the status of the vacuum breakers and are unnecessary. Therefore, the deletion of this requirement is acceptable.

3.6 3/4.7.F, Reactor Building - Suppression Chamber Vacuum Breakers

Proposed TS Section 3/4.7.F incorporates the requirements of STS Section 3/4.6.4.2 and current TS Section 3/4.7.A.3 for both stations. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR.

3.6.1 LCO

Current TS 3.7.A.3 is encompassed in proposed TS 3.7.F. The proposed LCO includes the provision that the vacuum breakers be closed. This requirement is not explicitly stated in the current TS. The proposed TS is an enhancement to the current TS by providing more explicit guidance and is therefore acceptable.

3.6.2 Applicability

The current TS requirement that the vacuum breakers be operable at all times when the primary containment is required, is equivalent to the proposed applicability of Modes 1, 2, and 3. Therefore, the proposed TS is consistent with current TS and is acceptable.

3.6.3 Required Actions

The current TS contain a requirement to lock closed an inoperable vacuum breaker. This requirement has a negligible safety benefit and has been deleted in the proposed TS. The proposed TS requires only that one vacuum breaker in the line be verified closed within 2 hours in the event it is inoperable. These valves also fall under the control of proposed TS Section 3.7.D, "Primary Containment Isolation Valves" which requires that an inoperable valve be secured in its isolated position. Therefore, the proposed TS provides an equivalent level of safety to ensure that primary containment

integrity is not violated. Therefore, this relaxation of current requirements is acceptable.

The proposed TS deviates from STS requirements in the AOT for an inoperable vacuum breaker. STS guidelines allow 72 hours while the proposed TS maintain the current AOT of 7 days. The proposed AOT is consistent with similar containment system requirements when one of a set of redundant components is inoperable. Because the proposed TS is consistent with current requirements, the deviation from STS guidelines is acceptable.

Proposed Action 3 is a new requirement not in current TS. Proposed Action 3 provides the requirements when the position indicator of the air operated reactor building - suppression chamber vacuum breaker is inoperable. The proposed action is consistent with STS guidelines, is an enhancement of current requirements, and is, therefore, acceptable.

3.6.4 Surveillance Requirements

Current TS 4.7.A.3.b requires an inspection of each vacuum breaker during each refueling outage. This requirement has not been retained in the proposed TS. These inspections are controlled by 10 CFR Part 50, Appendix J, testing and are not necessary to be included in the TS. The staff has determined that the requirements for vacuum breaker inspection 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 Part 50, Appendix J.

3.7 3/4.7.G, Drywell Internal Pressure

Proposed TS Section 3/4.7.G is a new section which incorporates the requirements of STS Section 3/4.6.1.6. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR.

3.7.1 LCO and Applicability

The proposed TS are based on STS guidelines. The proposed limit for drywell peak pressure is 1.5 psig and is consistent with the safety analyses for Dresden and Quad Cities. The proposed TS deviates from STS guidelines by including only a maximum pressure and not a minimum pressure requirement. The minimum pressure limit in the STS guidelines ensures that the external pressure differential doesn't exceed the design maximum external pressure differential. This requirement is unnecessary in the Dresden and Quad Cities TS due to the design of the vacuum breakers. The drywell to suppression chamber vacuum breakers open when the pressure in the drywell drops 0.5 psig below that in the suppression chamber. Reactor building to suppression chamber vacuum breakers maintain a minimum suppression chamber pressure by opening when the differential pressure is 0.5 psid. Therefore, the vacuum breaker design ensures that the pressure differential between the drywell and the reactor building does not exceed 1.0 psid and the minimum pressure requirements are not necessary. The proposed TS deviate from STS guidelines by providing an additional lower limit of 1.0 psig for the drywell during

operation above 15 percent power due to assumptions made in the hydrodynamic loading analysis performed for both Dresden and Quad Cities. This requirement adds additional restrictions which are necessary based on plant design and is acceptable. The proposed requirements are applicable to the plant design and provide enhanced guidance to operators. Therefore, the proposed TS is acceptable.

3.7.2 Required Actions

The proposed actions are adopted from STS guidelines and are applicable to Dresden and Quad Cities stations. The proposed requirements are an enhancement of current TS which contain no required actions for drywell pressure. Therefore, the proposed TS is acceptable.

3.7.3 Surveillance Requirements

The proposed TS requires measurement of drywell internal pressure every 12 hours. The proposed requirements are consistent with STS guidelines and applicable to the plant design. Therefore, the proposed TS is acceptable.

3.8 3/4.7.H, Drywell - Suppression Chamber Differential Pressure

Proposed TS Section 3/4.7.H incorporates the requirements of STS Section 3/4.6.2.4 and current TS Section 3/4.7.A.7 for Dresden, 3/4.7.A.6 for Quad Cities, and 3/4.2.E for both stations. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR.

3.8.1 Applicability

The proposed applicability is Mode 1, beginning 24 hours after thermal power is greater than 15 percent and ending 24 hours prior to reducing power below 15 percent. The 24 hours during power ascensions and descensions is provided in order to restore oxygen concentration to within limits and is consistent with STS guidelines. The current TS are applicable in the run mode which is approximately equal to 15 percent of rated thermal power. The proposed TS enhances the current TS by providing explicit requirements for power ascensions and planned power descensions. Therefore, the proposed applicability is acceptable.

3.8.2 LCO

The proposed TS revises the current minimum differential pressure limit for Quad Cities of 1.20 psid to 1.0 psid. The current limit for Dresden is 1.0 psid and, therefore, there is no change for Dresden. The purpose of maintaining the drywell at a slightly higher pressure, with respect to the suppression chamber, is to minimize the drywell pressure increase necessary to clear the downcomer pipes to commence condensation of steam in the suppression pool and to minimize the mass of the accelerated water leg. This reduces the hydrodynamic loads on the torus during the loss-of-coolant accident (LOCA) blowdown. Initial drywell-to-suppression chamber differential pressure affects both the pool dynamic loads on the suppression chamber and peak drywell pressure during clearing of the downcomer pipe during a design

operation above 15 percent power due to assumptions made in the hydrodynamic loading analysis performed for both Dresden and Quad Cities. This requirement adds additional restrictions which are necessary based on plant design and is acceptable. The proposed requirements are applicable to the plant design and provide enhanced guidance to operators. Therefore, the proposed TS is acceptable.

3.7.2 Required Actions

The proposed actions are adopted from STS guidelines and are applicable to Dresden and Quad Cities stations. The proposed requirements are an enhancement of current TS which contain no required actions for drywell pressure. Therefore, the proposed TS is acceptable.

3.7.3 Surveillance Requirements

The proposed TS requires measurement of drywell internal pressure every 12 hours. The proposed requirements are consistent with STS guidelines and applicable to the plant design. Therefore, the proposed TS is acceptable.

3.8 3/4.7.H, Drywell - Suppression Chamber Differential Pressure

Proposed TS Section 3/4.7.H incorporates the requirements of STS Section 3/4.6.2.4 and current TS Section 3/4.7.A.7 for Dresden, 3/4.7.A.6 for Quad Cities, and 3/4.2.E for both stations. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR.

3.8.1 Applicability

The proposed applicability is Mode 1, beginning 24 hours after thermal power is greater than 15 percent and ending 24 hours prior to reducing power below 15 percent. The 24 hours during power ascensions and descensions is provided in order to restore oxygen concentration to within limits and is consistent with STS guidelines. The current TS are applicable in the run mode which is approximately equal to 15 percent of rated thermal power. The proposed TS enhances the current TS by providing explicit requirements for power ascensions and planned power descensions. Therefore, the proposed applicability is acceptable.

3.8.2 LCO

The proposed TS revises the current minimum differential pressure limit for Quad Cities of 1.20 psid to 1.0 psid. The current limit for Dresden is 1.0 psid and, therefore, there is no change for Dresden. The purpose of maintaining the drywell at a slightly higher pressure, with respect to the suppression chamber, is to minimize the drywell pressure increase necessary to clear the downcomer pipes to commence condensation of steam in the suppression pool and to minimize the mass of the accelerated water leg. This reduces the hydrodynamic loads on the torus during the loss-of-coolant accident (LOCA) blowdown. Initial drywell-to-suppression chamber differential pressure affects both the pool dynamic loads on the suppression chamber and peak drywell pressure during clearing of the downcomer pipe during a design basis

accident. Drywell-to-suppression chamber differential pressure must be maintained within the specified limits so that the safety analysis remains valid. The limit of 1.0 psid was determined during the Mark I short-term program to provide the required safety margin in the suppression chamber design. The drywell to suppression chamber differential pressure is currently maintained at 1.20 psid at Quad Cities. The current safety analysis for Quad Cities specifies a minimum differential pressure of 1.0. Therefore, maintaining the dP at 1.20 provides additional margin but the dP may be reduced to 1.0 and the safety analysis would still apply. Therefore, the proposed change from Quad Cities current TS does not affect existing plant safety margins and is acceptable.

3.8.3 Required Actions

Proposed Action 1 requires that with the differential pressure less than the required limit, a 24 hour AOT is allowed and then power must be reduced below 15 percent within the next 8 hours. The 24 hour AOT deviates from STS guidelines of 8 hours, but is consistent with current requirements. The proposed 8 hour period to reach 15 percent power is an extension of two hours from the current TS. The 8 hour time period is consistent with STS guidelines and the additional 2 hours is not safety significant. The proposed action deviates from current TS which require that the plant be taken to cold shutdown if the LCO can not be met within the AOT. The current TS requires that the plant be in startup (below 15 percent power) within 6 hours and cold shutdown within the following 24 hours. The current requirement to bring the plant to cold shutdown conditions places an unnecessary thermal transient on the reactor vessel. The proposed TS places the reactor in a safe condition by placing it into a mode of operation for which the LCO no longer applies (below 15 percent of rated thermal power). Based on the above discussion, the proposed action is acceptable.

Proposed Actions 2, 3, and 4 are new requirements not in current TS. They provide the requirements for inoperable pressure instrumentation channels. The proposed actions deviate from STS guidelines because the system design at Dresden and Quad Cities prevents the literal adoption of STS guidelines. The Dresden and Quad Cities design only provides one direct suppression chamber to drywell differential pressure instrumentation channel. However, any pair of the redundant drywell and suppression chamber pressure instrumentation channels are sufficient to determine the differential pressure. Therefore, the proposed TS allows the drywell - suppression chamber differential pressure instrumentation channel, or the drywell and/or suppression chamber pressure instrumentation channel, to be inoperable for 30 days. With both inoperable, one method of determining differential pressure must be restored within 8 hours. These AOTs are consistent with STS guidelines. If the AOTs are exceeded, STS guidelines require that the plant be brought to cold shutdown. The proposed TS only requires that power be reduced below 15 percent power. This avoids unnecessary cycling of the reactor vessel and still brings the plant to a safe condition. The proposed TS is an enhancement of the current TS by ensuring that appropriate instrumentation is available to adequately measure drywell - suppression chamber differential pressure. Therefore, the proposed actions are acceptable.

3.8.4 Surveillance Requirements

The proposed TS relaxes the current periodicity of the SR from once per shift (8 hours) to every 12 hours, consistent with STS guidelines. However, the proposed SR expands on the current requirement of only recording the differential pressure by requiring an operability demonstration that the pressure is within limits. The additional requirements compensate for the decreased test frequency and the proposed TS is acceptable.

The proposed TS also includes requirements for a differential pressure instrumentation calibration once per 18 months and a channel check every 24 hours. The channel check is consistent with current Dresden TS and is an enhancement of current Quad Cities TS which do not require a channel check. The requirement for an instrument calibration every 18 months is a relaxation of current TS 3.2.E which requires a calibration every 6 months. An 18 month calibration frequency is consistent with STS guidelines. The proposed frequency is acceptable for Quad Cities based on historical calibration data which demonstrates that instrument drift is insignificant for periods of less than 18 months. The licensee had not evaluated the historical data for Dresden at the time of the initial submittals. Therefore, this item will remain an open item for Dresden pending the licensee's evaluation and will be addressed in the clean-up amendment.

The proposed TS does not adopt the STS guideline for a low differential pressure alarm setpoint because there are no similar current TS requirements. Therefore, there is no reduction in existing plant safety margins and this deviation is acceptable.

3.9 3/4.7.I, Primary Containment Nitrogen Purge System

Proposed TS Section 3/4.7.I incorporates the requirements of STS Section 3/4.6.2 and current Dresden TS Section 3/4.7.A.6. There are no comparable current TS requirements for Quad Cities. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR.

3.9.1 Applicability

The proposed applicability of Modes 1 and 2 encompasses the current applicability of power operation and is consistent with STS guidelines. Therefore, the proposed applicability is acceptable.

3.9.2 LCO

The proposed LCO maintains the current requirement of an operable primary containment nitrogen system and enhances the current TS by specifying that an operable inerting flow path and an operable make-up flow path are required. Therefore, the proposed LCO is acceptable.

Current Dresden TS 3.7.A.6.e which specifies a maximum containment repressurization pressure using the containment makeup inerting system has not been retained in the proposed TS. This information is a design detail which

is not required to be included in TS. The specific methodology for assuring the LCO is maintained is a plant design detail which is controlled by station emergency operating procedures. The staff has determined that the requirements for a maximum repressurization pressure 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 proposed change is acceptable.

3.9.3 Required Actions

The current TS include a 7 day AOT if the LCO can not be met. The September 17, 1993, submittal proposed a 30 day AOT. The required action will remain as an open item contingent upon its disposition in the TSUP cleanup amendment.

3.9.4 Surveillance Requirements

The current requirement to record the liquid nitrogen storage tank level weekly has been relaxed to monthly in the proposed TS based on STS guidelines. A monthly verification of storage tank level has been shown based on industry experience to adequately maintain the safety margin for system operability. In addition, the required minimum level in the storage tanks has been increased from 60 inches in the current TS to 70 inches. This requirement is more conservative than the current TS and provides greater assurance that sufficient nitrogen will be available. The proposed requirements do not significantly reduce existing plant safety margins and are acceptable.

The current Dresden TS require checking nitrogen storage tank level after reinerting containment. This requirement has not been retained in the proposed TS. This requirement is encompassed within the requirements in TS 3.0.D which requires that all conditions required in the future mode are satisfied prior to a change of mode. Therefore, the requirement for a minimum tank level must be satisfied after reinerting containment and it is unnecessary to repeat this requirement in TS section 4.7. The current Quad Cities TS do not contain this requirement and therefore there is no change in the proposed TS.

The current Dresden TS require monthly valve actuation to determine operability. This requirement has not been retained in the proposed TS. Instead of valve actuation, the proposed TS require nitrogen tank quantity measurement and verification of proper system valve positioning every 31 days. The proposed TS are based on STS guidelines which has been shown based on industry experience to provide an adequate level of protection for ensuring the operational readiness of the system. The actuation of system valves to demonstrate operability is inconsistent with the intent of GL 93-05 to reduce unnecessary system actuations to demonstrate system performance. A verification of valve positioning will ensure proper system readiness. Therefore, the deletion of this requirement is acceptable. The current Quad Cities TS do not require valve actuation, therefore there is no change in the proposed TS.

3.10 3/4.7.J, Primary Containment Oxygen Concentration

Proposed TS Section 3/4.7.J incorporates the requirements of STS Section 3/4.6.6.4 and current TS Section 3/4.7.A.5 for both stations. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR.

3.10.1 LCO and Applicability

The current TS applicability is the run mode. The proposed applicability is Mode 1 above 15 percent power. The current definition of the run mode is Mode 1, which must be entered prior to increasing power above 15 percent of rated thermal power. The proposed TS do not apply until 24 hours after 15 percent power is reached and end 24 hours prior to reducing power below 15 percent. This 24 hour time period is provided in order to restore the oxygen concentration to within limits. The current Quad Cities TS specify these provisions for power ascensions and planned power descensions and, therefore, the proposed TS are consistent with the current Quad Cities TS. The current Dresden TS does not provide explicit requirements for power ascensions and planned power descensions. The proposed requirements provide enhanced guidance to appropriately define the conditions for which primary containment oxygen concentration is required. The proposed applicability provides an equivalent level of safety as compared with the current TS and is acceptable.

Current TS 3.7.A.6.d requires that the primary containment oxygen sampling system be operable to measure oxygen concentration. This requirement has been deleted in the proposed TS. The oxygen analyzer is controlled by proposed TS 4.7.J which requires oxygen concentration to be determined every 7 days. The specific requirements for operability of the system used to determine oxygen concentration are design details more appropriate for administrative controls. The current TS requirements for testing and calibration of the oxygen analyzing system have also been removed to administrative controls. The proposed TS will continue to assure that the primary containment oxygen concentration is maintained within appropriate levels and is acceptable. The staff has determined that the requirements for the primary containment oxygen sampling system are not required to be in the TS under 10 CFR 50.36 or Section 182a of the Atomic Energy Act. Further, they do not fall within any of the four criteria discussed in Section 2.0 above. Therefore, the deletion of this requirement is acceptable.

3.10.2 Required Actions

Current TS require that if the LCO is not met and the limits can not be restored within 24 hours, then the plant is required to be in startup/hot standby within the next 6 hours. The proposed TS extend the time to bring the plant to less than 15 percent power (equivalent to startup/hot standby) to 8 hours. This deviation does not significantly reduce existing plant safety margins as it provides a more reasonable time frame to place the reactor in the appropriate conditions. Therefore, the proposed action is acceptable.

3.10.3 Surveillance Requirements

The proposed SR requires verification of suppression chamber oxygen concentration once per 7 days in Mode 1 consistent with the current TS. The proposed TS enhance the current TS by expanding the required testing to include verification of oxygen concentration within 24 hours after thermal power reaches 15 percent. Therefore, the proposed SR is acceptable.

3.11 3/4.7.K, Suppression Chamber

Proposed TS Section 3/4.7.K incorporates the guidelines of STS Section 3/4.6.2 and current TS Section 3/4.7.A.1 for both stations. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR.

3.11.1 Applicability

The current TS specify applicability as "at any time that the nuclear system is pressurized above atmospheric." This is equivalent to the proposed applicability of Modes 1, 2, and 3. The current TS also specify applicability during work which has the potential to drain the vessel. This requirement has been relocated to proposed TS Section 3/4.5.C, "ECCS - Suppression Chamber," which will be evaluated in a separate amendment. This will remain an open item pending issuance of the amendments for TSUP section 3/4.5.

3.11.2 LCO

The current suppression pool water volume limits have been retained as equivalent water level limits in the proposed TS. The current Quad Cities TS specify that the maximum and minimum water volume are shown on the level indicator at ±2 inches. The level indicator readings have not been retained. This information provides the specific methodology for satisfying the performance of the surveillance requirement and is not necessary to be controlled in the TS. The staff has determined that these requirements are not required to be in the TS under 10 CFR 50.36 or Section 182a of the Atomic Energy Act and they do not fall within any of the four criteria discussed in Section 2.0 above. This deletion does not change any TS requirements and is, therefore, acceptable.

Current requirements regarding minimum and maximum downcomer submergence are not included in the proposed TS. The discussion of downcomer submergence is now included in the Bases section and describes the relationship of downcomer submergence and suppression pool level. There are no specific actions delineated in the current TS if these requirements are not met. The maximum and minimum suppression pool level allowed by proposed TS 3.7.K.1 results in a downcomer submergence equivalent to the current maximum and minimum downcomer submergence TS limit. Therefore, retention of the downcomer submergence limitations within the TS would be redundant to proposed TS 3.7.K.1. The staff has determined that these requirements are not required to be in the TS under 10 CFR 50.36 or Section 182a of the Atomic Energy Act and they do not fall within any of the four criteria discussed in Section 2.0 above. Therefore, the deletion of these requirements is acceptable.

Proposed TS 3.7.K.3 is based on current TS 4.7.A.4.b.4 which requires that a drywell to suppression chamber leak test be performed every 18 months. This SR has been converted to an LCO consistent with STS guidelines. The limits have not been changed. The proposed TS enhances current requirements by explicitly defining the requirement as an LCO with a required action. This change is acceptable.

3.11.3 Required Actions

Proposed Action 1 requires that if the suppression pool level is outside the allowed limits and can not be restored within 1 hour, the plant must be in hot shutdown in 12 hours and cold shutdown in the following 24 hours. The current Dresden TS require that if level can not be restored within 6 hours the reactor must be brought to cold shutdown in the following 24 hours. The proposed TS are a relaxation from 30 hours to 37 hours to reach cold shutdown. However, the proposed requirements ensure that actions are initiated earlier to bring the reactor to a safe condition and the additional 7 hours does not significantly affect the level of safety. The proposed TS is therefore acceptable. The current Quad Cities TS do not contain a similar required action. Therefore, the proposed action is an enhancement of current Quad Cities TS and is acceptable.

Proposed Actions 2 and 3 require that, if the suppression pool average water temperature is greater than 95 degrees Fahrenheit in Modes 1 or 2 or greater than 105 degrees Fahrenheit during testing that adds heat to the pool, after 24 hours thermal power must be reduced to less than or equal to one percent rated thermal power. Reducing power places the reactor in a safer condition since the current and proposed TS allow temperatures of up to 110 degrees Fahrenheit when power is less than 1 percent. The current TS require the temperature to be reduced to less than 95 degrees Fahrenheit without specifying a reduction in thermal power. The proposed TS maintains an equivalent level of safety as the current TS since the reactor is placed in a condition which complies with the TS. Once the plant is below one percent power, the suppression pool temperature can safely be maintained up to 110 degrees Fahrenheit. If the temperature is above 110 degrees Fahrenheit, proposed Action 4 requires that the reactor mode switch immediately be placed in the shutdown position. Proposed Action 4 is consistent with current TS and is acceptable. The proposed actions provide an equivalent level of protection as the current TS and are acceptable.

STS guidelines include actions for inoperable temperature elements for the suppression pool. Because the current TS do not provide requirements, they have not been incorporated into the proposed TS. Action requirements for suppression pool temperature monitoring will be included in proposed TS Section 3/4.2, "Instrumentation." The proposed TS are consistent with current TS and are, therefore, acceptable.

3.11.4 Surveillance Requirements

Current TS require the performance of continual water temperature monitoring whenever there is indication of relief valve operation. This requirement has not been retained in the proposed TS. The proposed TS requires that the

suppression pool water temperature be verified within limits during testing which adds heat to the suppression pool. The requirements specified in TSUP 3.6.F for relief valve operation (which was approved by amendment number 140 and 134 to the Dresden TS and amendments 162 and 158 to the Quad Cities TS) specify the required actions for open relief valves based on the suppression pool water temperature. This requirement, in addition to proposed TS 3.7.K will ensure that suppression chamber water level and temperature are maintained within limits. Therefore, it is not necessary to monitor water temperature whenever there is indication of relief valve operation. This change is acceptable.

The current TS also require continual monitoring of water temperature during testing which adds heat to the suppression pool. The proposed TS delete the requirement for continual monitoring and instead require verification of water temperature every 5 minutes. Requirements to continually monitor pool water temperature are not included in the STS guidelines. This requirement is accomplished by verification of temperature every 5 minutes which provides an equivalent level of protection based on industry experience. Therefore, the proposed SR is acceptable.

Current Dresden TS 4.7.A.2.g requires a visual inspection of the interior surfaces of the drywell each operating cycle. This requirement has been deleted in the proposed TS. The visual inspection of the drywell is considered a preventive maintenance activity and does not contribute to the operability of the drywell. Therefore, this inspection is more appropriately controlled in owner controlled documentation. The staff has determined that the requirement for a visual inspection of the drywell surface is not required to be in the TS under 10 CFR 50.36 or Section 182a of the Atomic Energy Act. Further, it does not fall within any of the four criteria discussed in Section 2.0 above. This requirement is not in the current Quad Cities TS or STS guidelines. Based on the above discussion, the deletion of this requirement is acceptable.

A new requirement, proposed TS 4.7.K.4, was added which requires a visual inspection of the accessible interior and exterior surfaces of the suppression chamber. This TS is adopted from STS guidelines. The proposed TS is an enhancement of current TS and is acceptable.

Proposed TS 4.7.K.5 requires a drywell to suppression chamber bypass leakage test. The proposed TS enhances the current TS by adding the requirement that if the test fails to meet the limit, tests must be performed more frequently. This is an enhancement of current TS and is acceptable.

3.12 <u>3/4.7.L. Suppression Chamber and Drywell Spray</u>

Proposed TS Section 3/4.7.L incorporates the requirements of STS Section 3/4.6.6.2 and current TS Section 3.5.A.6 for Dresden and 3.5.B.4 for Quad Cities. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR.

3.12.1 Applicability

The proposed applicability of Modes 1, 2, and 3 is consistent with the current requirement of "all times when the reactor coolant temperature is greater than 212 degrees Fahrenheit." Therefore, the proposed applicability is acceptable.

3.12.2 LCO

The proposed LCO is based on STS guidelines. The proposed requirements deviate from STS guidelines for residual heat removal (RHR) due to plant design limitations at Dresden Station which has no RHR system. Therefore, the low-pressure coolant injection (LPCI)/containment cooling terminology is incorporated for the Dresden LCO. There are no current LCO requirements for Dresden or Quad Cities, therefore, the proposed LCO is an enhancement and is acceptable.

3.12.3 Required Actions

Proposed Action 1 requires that with one suppression chamber/drywell spray loop inoperable, the inoperable loop must be restored within 7 days. The current TS include a 30 day AOT for one drywell spray loop inoperable. Therefore, the proposed TS are an enhancement for drywell spray. The current TS do not provide an action for an inoperable suppression chamber spray loop. The proposed TS are an enhancement of current TS and are acceptable.

Proposed Action 2 includes an 8 hour AOT when both loops are inoperable. There are no comparable current TS requirements so the current TS would default to TS 3.0.A which requires cold shutdown in 24 hours with no AOT. This relaxation of current requirements does not significantly reduce existing plant safety margins as an 8 hour AOT provides a reasonable time to restore the system and possibly avoid an unnecessary reactor transient and shutdown. Therefore, the proposed action is acceptable.

Proposed Actions 1 and 2 require that, at the conclusion of the AOT, the plant must be brought to hot shutdown within 12 hours and cold shutdown within the following 24 hours. This is a relaxation of the current TS which require cold shutdown within 24 hours. The proposed TS ensure that action is taken sooner to place the plant in a safe condition but allow more time for a controlled shutdown to cold shutdown conditions. Based on industry experience, the proposed actions provide an adequate level of safety. Therefore, the proposed requirements are acceptable.

Proposed Action 2 in the Quad Cities TS contains a footnote which states that if cold shutdown cannot be attained whenever two RHR shutdown cooling mode subsystems are inoperable, then reactor coolant temperature must be maintained as low as practical by use of alternate heat removal methods. This note is adopted from STS guidelines and is necessary due to Quad Cities design. The proposed Dresden TS do not contain this footnote because the system used for this function is not related to the ability to achieve cold shutdown.

3.12.4 Surveillance Requirements

Proposed TS 4.7.L.1 is consistent with the current SR for Dresden and STS guidelines. There is not a comparable SR in the current Quad Cities TS. Therefore, the proposed SR is equivalent to the current Dresden TS and an enhancement of the current Quad Cities TS and is acceptable.

Proposed TS 4.7.L.2 is based on current SRs for both stations, but adds the option of a smoke test in addition to the air test to satisfy the SR. This change is consistent with current industry practices for drywell spray nozzle testing which are applicable to Dresden and Quad Cities. The smoke test provides an equivalent test to satisfy the SR. The proposed TS eliminates the requirement to test the drywell spray header in accordance with STS guidelines. However, the proposed TS contain a new requirement to verify that the spray nozzle is unobstructed. The proposed TS will provide an equivalent level of safety and is, therefore, acceptable.

The current Quad Cities TS contain a requirement to perform a water spray test of the torus spray header and nozzle every five years. The current Dresden TS do not contain this requirement. Because flow through the suppression pool spray nozzles could potentially damage equipment by spraying in the suppression chamber, the licensee originally proposed eliminating this test to reduce potential equipment damage. Based on discussions with the NRC staff, the licensee will revise the proposed TS to include a test of suppression chamber spray. This will remain an open item pending its approval in the clean-up amendment.

3.13 3/4.7.M, Suppression Pool Cooling

Proposed TS Section 3/4.7.M incorporates the guidelines of STS Section 3/4.6.2.3 and current TS Section 3/4.5.B for both stations. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR.

3.13.1 Applicability

The proposed applicability of Modes 1, 2, and 3 is equivalent to the current requirement at Dresden which states "when the reactor coolant temperature is greater than 212 degrees Fahrenheit" and the current requirement at Quad Cities which is "prior to reactor startup from a cold shutdown." Therefore, the proposed applicability is acceptable.

3.13.2 LCO

The proposed LCO requirement that both loops be operable is consistent with the current TS requirement and is acceptable.

3.13.3 Required Actions

Proposed Actions 1 and 2 require that if the LCO can not be met the plant must be in hot shutdown in 12 hours and cold shutdown within the following 24 hours. This is a deviation from the current TS which require cold shutdown in 24 hours. The proposed TS provide an equivalent level of protection since they require the plant to be placed in a safer condition in a more expeditious time frame. The extension of 12 hours to reach cold shutdown conditions does not have a significant impact on plant safety. Therefore, the proposed TS are acceptable.

3.13.4 Surveillance Requirements

Proposed TS 4.7.M.1 is consistent with the current Dresden SR and STS guidelines. There are no comparable SRs in the current Quad Cities; therefore, this TS is an enhancement of the current TS and is acceptable.

Proposed TS 4.7.M.2 requires that pump flow be evaluated under the inservice test program. The current TS specify the requirements for the containment cooling pumps to demonstrate operability. Proposed TS 4.7.M.2 changes the required test frequencies to "pursuant to Specification 4.0.E." TS 4.0.E states that SRs for inservice testing be performed at the frequency specified by Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code except where relief has been granted by the Commission. The proposed TS does not change any requirements but rather, indicates that the test frequency will be in accordance with the IST program. Revisions to the IST program are controlled by the requirements of 10 CFR 50.55a. The staff has determined that these requirements are not required to be in the TS under 10 CFR 50.36 or Section 182a of the Atomic Energy Act. Further, they do not fall within any of the four criteria discussed in Section 2.0 above. In addition, the staff finds that sufficient regulatory controls exist under 10 CFR 50.55a. The staff finds the proposed TS acceptable.

3.14 3/4.7.N, Secondary Containment Integrity

Proposed TS Section 3/4.7.N incorporates the requirements of STS Section 3/4.6.5 and current TS Section 3/4.7.C for both stations. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR.

3.14.1 Applicability

The proposed applicability is during Modes 1, 2, and 3, and when handling irradiated fuel in the secondary containment, during core alterations, and during operations with potential for draining the vessel. The proposed applicability encompasses the current TS which require secondary containment at all times with four exceptions. When all four exceptions are applicable, the resulting applicability is consistent with the proposed applicability. However, the exception regarding when the reactor coolant system is vented is not encompassed within the proposed TS. Current TS would require secondary

containment to be operable in cold shutdown unless the reactor coolant system were vented. The proposed TS would not require secondary containment to be operable when the plants are in cold shutdown without the reactor vented. This change is acceptable because in cold shutdown the reactors do not achieve high enough temperature to steam and, therefore, can not pressurize the reactor by nuclear sources such that secondary containment would be required.

3.14.2 LCO

The current TS contain a requirement that the doors of the core spray and RHR pump compartments be closed at all times except during passage in order to consider these systems operable. This requirement is a design detail which defines the operability of these systems and has not been retained in the proposed TS. This requirement is more appropriate for administrative control The staff has determined that this requirement is not required to be in the TS under 10 CFR 50.36 or Section 182a of the Atomic Energy Act and does not fall within any of the four criteria discussed in Section 2.0 above. Therefore, its deletion from TS is acceptable.

3.14.3 Required Actions

Proposed Action 1 requires that without secondary containment integrity in Modes 1, 2, or 3, integrity must be restored within 4 hours or the plant must be in hot shutdown in 12 hours and cold shutdown in the following 24 hours. This is equivalent to the current TS at Dresden. There are no comparable current TS requirements for Quad Cities; therefore, TS 3.0.A applies and the plant must be in cold shutdown in 24 hours. The proposed requirements (16 hours to bring the plant to hot shutdown) provide an equivalent level of protection for ensuring appropriate controls are enforced as compared to current TS (24 hours to bring the plant to cold shutdown). The proposed TS provide enhanced operator guidance to the Quad Cities TS and are consistent with the current Dresden TS. Therefore, the proposed TS is acceptable.

3.14.4 Surveillance Requirements

The current TS for Quad Cities that require a preoperational test and additional tests during the first operating cycle have not been retained in the proposed TS since these requirements are obsolete. This change is acceptable.

The current TS contain requirements that the doors of the core spray and RHR pump compartments be closed at all times except during passage in order for those systems to be considered operable. These requirements are design details regarding the flood protection requirements for the affected system and are not necessary to be included in the TS. The definition of operability in TS Section 3/4.0 is sufficient. The specific details related to the status of the compartment doors are more appropriate for administrative control. The staff has determined that this requirement is not required to be in the TS under 10 CFR 50.36 or Section 182a of the Atomic Energy Act and it does not

fall within any of the four criteria discussed in Section 2.0 above. Therefore, this change is acceptable.

The proposed TS add two new SRs based on STS guidelines which enhance the integrity of secondary containment. This is an enhancement of the current TS and is, therefore, acceptable.

3.15 3/4.7.0, Secondary Containment Automatic Isolation Dampers

Proposed TS Section 3/4.7.0 is a new section which incorporates the guidelines of STS Section 3/4.6.5.2. There are no current TS requirements for the secondary containment automatic isolation dampers for Dresden or Quad Cities. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR.

3.15.1 Applicability

The proposed applicability is during Modes 1, 2, and 3, and when handling irradiated fuel in the secondary containment, during core alterations, and operations with potential for draining the reactor vessel. The proposed applicability is consistent with STS guidelines and applicable to the Dresden and Quad Cities design. The proposed applicability is an enhancement of current TS and is therefore acceptable.

3.15.2 LCO

The proposed LCO ensures that the secondary containment automatic isolation dampers, in combination with other accident mitigation systems, limit fission product release during and following a postulated design basis accident such that offsite radiation exposures are maintained within the requirements of 10 CFR Part 100. The proposed LCO is based on STS guidelines with the following exceptions. The proposed TS do not incorporate STS Table 3.6.5.2-1 which lists each valve that is controlled by the TS. This change is consistent with the guidance in GL 91-08. The proposed TS also does not require that the 18 month actuation test be performed during shutdown in accordance with the guidance of GL 91-04. The proposed TS do not specify maximum isolation times since no specific reactor building ventilation system isolation times are assumed in the existing safety analysis. The proposed TS is an enhancement of the current TS which contain no current requirements and the deviations from STS guidelines are acceptable. Therefore, the proposed LCO is acceptable.

3.15.3 Surveillance Requirements

The proposed SRs are based on STS guidelines except that the requirements regarding specific isolation times are not adopted. This change is acceptable because no specific reactor building ventilation system isolation times are assumed in the existing safety analysis. The proposed TS is an enhancement of current TS which contain no requirements and is, therefore, acceptable.

3.16 3/4.7.P, Standby Gas Treatment System

Proposed TS Section 3/4.7.P incorporates the requirements of STS Section 3/4.6.5.3 and current TS Section 3/4.7.B for both stations. Plant specific values for the listed parameters are included to be consistent with each station's UFSAR.

3.16.1 Applicability

The proposed applicability is during Modes 1, 2, and 3, and when handling irradiated fuel in the secondary containment, during core alterations, and operations with potential for draining the reactor vessel. This is consistent with the current requirement of whenever secondary containment is required (Modes 1, 2, and 3, when handling irradiated fuel in the secondary containment, and during operations with the potential for draining the vessel). Therefore, the proposed applicability is acceptable.

3.16.2 LCO

The current LCO requires that two standby gas treatment subsystems be operable. The proposed TS maintain this requirement and expand it to include an operable diesel generator power source. The proposed TS enhances current requirements and is, therefore, acceptable.

3.16.3 Required Actions

Proposed Actions 2 and 3 contain the requirements for both subsystems of the standby gas treatment system (SGTS) inoperable due to inoperable diesel generators or a combination of an inoperable diesel generator and the other subsystem of SGTS inoperable for other reasons. In these circumstances, the proposed TS allows a 72 hour AOT after which time the plant must be in hot shutdown within 12 hours and cold shutdown within 24 hours. The proposed action statement is a significant relaxation of the current TS which require that the plant be in cold shutdown in 36 hours. The current TS do not provide an exception for inoperability due to inoperable diesel generators. Proposed Action 2 and 3 are not acceptable. This will remain an open item pending submittal of revised proposed Actions by the licensee and approval in the clean-up amendment.

Proposed Action 4 discusses the requirements for both subsystems inoperable for reasons other than inoperable diesel generators. The proposed Action is to restore one subsystem within one hour or be in hot shutdown within 12 hours and cold shutdown within the following 24 hours. The proposed Action is consistent with the current requirement (cold shutdown in 36 hours) except that it includes a one hour AOT. The proposed Action is consistent with STS guidelines. The additional hour is a reasonable period of time to attempt to restore one subsystem before initiating a shutdown and does not affect the existing margin of safety. In addition, the proposed Action enhances the current TS by providing a specified time in which the plant must reach hot

shutdown, ensuring that the plant is in a more conservative mode in a shorter period of time. The proposed Action is acceptable.

The current Quad Cities TS contain a requirement to perform an operability demonstration of a redundant SGTS subsystem whenever the opposite train of SGTS is inoperable. This requirement has been deleted in the proposed TS. The original need for demonstrating operability of redundant subsystems was based on a lack of industry plant operating history and equipment failure data. However, plant operating history now shows that testing of redundant systems when one system is inoperable is not necessary to provide adequate assurance of system operability. In addition, removal of the redundant system from service for testing increased the risk that the redundant system will fail. Actual observations of this configuration have indicated that failures of the redundant system are related to the testing itself, and are not an indication that the system would have failed should it have been needed. Operability of these subsystems can be verified through an administrative check of valve lineups, electrical lineups and instrumentation requirements. If these have not changed since the most recent verification of operability, then the subsystem can be considered operable. The staff has determined that the requirement to perform operability demonstrations for the redundant SGTS subsystem is not required to be in the TS under 10 CFR 50.36 or Section 182a of the Atomic Energy Act. Further, is does not fall within any of the four criteria discussed in Section 2.0 above. Therefore, the deletion of this requirement is acceptable.

3.16.4 Surveillance Requirements

The proposed TS incorporates revised laboratory test conditions for the laboratory sample analysis of charcoal canisters. The current TS require testing at 130 degrees Celsius and 95 percent relative humidity. The proposed TS specify 30 degrees Celsius and 70 percent relative humidity. These changes implement the applicable requirements of the American Society for Testing and Materials (ASTM) D-3803-89, "Standard Test Method for Nuclear-Grade Activated Carbon" and the actual operating configuration for the SBGT system. The current laboratory test condition requirement for the analysis of activated carbon are specified to approximate operating or accident conditions which would severely reduce the performance of activated carbon. The current values of 130 degrees Celsius and 95 percent relative humidity are not realistic and potentially non-conservative. While testing at 95 percent relative humidity is conservative for an atmospheric cleanup system with an inlet heater, the effects are offset when testing at 130 degrees Celsius. The higher temperature utilized during the equalization portion of the activated carbon test has a regenerative effect on the charcoal (i.e., the adsorbed radioactive materials are driven off/released, which leads to falsely conservative values for charcoal efficiency). The SGTS is equipped with an electric inlet heater, which is designed to reduce relative humidity to 70 percent at the inlet of the charcoal adsorbers. The current requirements for temperature resulted from an inadequate design review of the initial control room habitability study and applicable standards. The proposed SR specifies more realistic test conditions and is acceptable.

The current TS require a once-per-cycle demonstration of the SGTS inlet heater operability at rated power but doesn't specify an acceptable value for rated power. The proposed TS adopts an acceptance criteria based on the rated capacity of the heater. This is consistent with STS guidelines and provides a method to accurately verify the ability of the inlet heater to reduce the relative humidity of inlet air to less than 70 percent. In addition, the proposed TS establishes a voltage correction to the rated capacity acceptance criteria. Allowing supply voltage to be corrected for expected variations from the nominal value is acceptable as these requirements do not affect the ability of the filter inlet heaters to reduce the relative humidity below the TS limit. The proposed TS is based on the design capability of the filter inlet heater (30 $\pm$ 3.0 kW) that is readily measurable with available test equipment. The proposed TS removes ambiguities associated with current acceptance criteria for heater performance and potential degradation and do not alter the ability of the inlet heaters to perform their design function.

The current Dresden TS require that once per 720 hours of SGTS operation, a performance test of the charcoal adsorber must be performed. The proposed TS extends the usage requirement to 1440 hours. The proposed TS is consistent with current Quad Cities TS which already have a 1440 hour requirement. The staff has determined that an extension to 1440 hours for Dresden is not conservative and this change is unacceptable. This issue will remain an open item pending resolution and approval in the clean-up amendment.

3.17 Bases

The staff has reviewed the proposed Bases for TS 3/4.7. The proposed Bases have been prepared using the guidelines of the STS. The staff finds these proposed Bases acceptable.

3.18 Open Items

The following item should be left as an open item, contingent upon correction in the TSUP clean-up amendment.

1. Frequency of the channel calibration in TS 4.7.H.2.b for Dresden
2. The allowed outage time of TS 3.7.I, Action
3. The applicability of the suppression chamber TS during work which has the potential to drain the vessel will be relocated to proposed TS 3.5.C.
4. A spray test of the suppression chamber every five years should be included in TS 3.7.L.
5. TS 3.7.P, Actions 2 and 3
6. TS 4.7.P.3, the hours of charcoal adsorber operation after which an analysis of a carbon sample is required

4.0 SUMMARY

The proposed TS for Section 3/4.7 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 this amendment do not pose any decrease in safety, or an increase in the probability of an analyzed or unanalyzed accident. The revised TS changes do not reduce the existing margin of safety set forth by the current TS. Therefore, the staff finds the proposed TS changes, with the exception of the open items, acceptable.

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (60 FR 39433). 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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