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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 125
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 June 1, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 125, 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 within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

James E. Dyer

James E. Dyer, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 11, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 125

FACILITY OPERATING LICENSE NO. DPR-19

DOCKET NO. 50-237

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 and contain marginal lines indicating the area of change. Pages indicated with an asterisk were inadvertently deleted in Amendment No. 122.

REMOVE	INSERT
iii	iii
v	v
1.0-3	1.0-3
3/4.7-7	3/4.7-7
3/4.7-9	3/4.7-9
3/4.7-19	3/4.7-19
-	3/4.7-19a
-	3/4.7-19b
-	*3/4.7-31
-	*3/4.7-32
B 3/4.7-33	B 3/4.7-33
B 3/4.7-34	B 3/4.7-34
B 3/4.7-43	B 3/4.7-43
-	B 3/4.7-43a

(Table of Contents, Cont'd.)

(Table of Contents, Cont'd.)

1.0 DEFINITIONS (Cont'd)

R. Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All manual containment isolation valves on lines connecting to the reactor coolant system or containment which are not required to be open during accident conditions are closed, or comply with the requirements of Specification 3.7.D.

2. Each primary containment air lock is in compliance with the requirements of Specification 3.7.A.8.

3. All automatic containment isolation valves are operable or deactivated in the isolated position, or comply with the requirements of Specification 3.7.D.

4. All blind flanges and manways are closed.

S. Protective Instrumentation Definitions

1. Instrument Channel - An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel.

2. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

3. Protective Action - An action initiated by the protection system when a limit is reached. A protective action can be at a channel or system level.

4. Protective Function - A system protective action which results from the protective action of the channels monitoring a particular plant condition.

T. Rated Neutron Flux - Rated neutron flux is the neutron flux that corresponds to a steady-state power level of 2527 thermal megawatts.

U. Rated Thermal Power - Rated thermal power means a steady-state power level of 2527 thermal megawatts.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

(b) Deleted

(c) 11.5 SCF per hour for any main steam isolation valve at a test pressure of 25 psig.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

c. If two consecutive Type A tests fail to meet either 75 percent of L_a or 75 percent of L_t, a Type A test shall be performed at each shutdown for refueling or approximately every 18 months until two consecutive Type A tests meet the above requirements, at which time the normal test schedule may be resumed.

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

3.7 LIMITING CONDITION FOR OPERATION
(Cont'd.)

4.7 SURVEILLANCE REQUIREMENTS
(Cont'd.)

(1) Main steam line isolation valves which shall be tested at a pressure of 25 psig each operating cycle.

(2) Bolted double-gasketed seals which shall be tested at a pressure of 48 psig whenever the seal is closed after being opened and each operating cycle.

(3) Air locks shall be tested per Specification 4.7.A.8.

(4) Deleted

f. Continuous Leak Rate Monitor

(1) When the primary containment is inerted, the containment

3.7 LIMITING CONDITION FOR OPERATION (cont'd)

shall be initiated and the reactor shall be in a cold shutdown condition in the following 24 hours.

8. Primary Containment Air Locks

a. 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 airlock leakage rate of less than or equal to 0.05 La at Pa, 48 psig.

b. With one primary containment air lock door inoperable:

(1) Maintain at least the operable air lock door closed[a] and either restore the inoperable air lock door to operable status within 24 hours or lock the operable air lock door closed.

4.7 SURVEILLANCE REQUIREMENTS (cont'd)

8. Primary Containment Air Locks

a. Each primary containment air lock shall be demonstrated operable:

(1) By conducting an overall air lock leakage test at Pa, 48 psig and verifying that the overall air lock leakage rate is within its limit:

(a) Within 72 hours of air lock opening when 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[b], and

(c) Prior to establishing Primary Containment Integrity following air lock opening.

a 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.

b The provisions of Specification 1.0.CC are not applicable.

3.7 LIMITING CONDITION FOR OPERATION (cont'd)

(2) 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[a] at least once per 31 days.

(3) Otherwise, be in at least hot shutdown within the next 12 hours and in cold shutdown within the following 24 hours.

c. With the primary containment air interlock mechanism inoperable:

(1) Operations may continue provided the air lock is otherwise operable and entry and exit of the primary containment is administratively controlled by a dedicated individual.

(2) Otherwise, restore the air lock interlock mechanism to operable status within 24 hours or lock the operable air lock door closed and verify that the operable air lock door is locked closed at least once per 31 days.

4.7 SURVEILLANCE REQUIREMENTS (cont'd)

(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 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.

3.7 LIMITING CONDITION FOR OPERATION (cont'd)

d. With the primary containment air lock inoperable, except as a result of an inoperable air lock door or air lock interlock mechanism:

(1) Maintain at least one air lock door closed.

(2) 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 at least cold shutdown within the following 24 hours.

B. Standby Gas Treatment System

1. Two separate and independent standby gas treatment system subsystems shall be operable at all times when secondary containment integrity is required, except as specified in sections 3.7.B.1(a) and (b).

a. After one of the standby gas treatment system subsystems is made or found to be inoperable for any reason, reactor operation and fuel handling is permissible only during the succeeding seven days, provided that all active components in the other standby gas treatment subsystem shall be operable.

4.7 SURVEILLANCE REQUIREMENTS (cont'd)

B. Standby Gas Treatment System

1. At least once per month, initiate from the control room 4000 cfm (plus or minus 10%) flow through each subsystem of standby gas treatment system for at least 10 hours with the subsystem heaters operating at rated power.

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3.7 LIMITING CONDITION FOR OPERATION BASES

A. Primary Containment - The integrity of the primary containment and operation of the emergency core cooling system in combination, limit the off-site doses to values less than those suggested in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception is made to this requirement during initial core loading and while the low power test program is being conducted during initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required. There will be no pressure on the system at this time which will greatly reduce the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth to preclude a peak fuel enthalpy of 280 cal/gm. In addition, in the unlikely event that an excursion did occur, the reactor building and standby gas treatment system, which shall be operational during this time, offers a significant barrier to keep off-site doses well within 10 CFR 100.

The primary containment air lock's structural integrity and leak tightness are essential to the successful mitigation of a design basis accident event (DBA). The air lock is required to be operable whenever primary containment integrity is required. For the air lock to be considered operable, the air lock interlock mechanism must be operable, the air lock must be in compliance with the 10 CFR 50, Appendix J, Type B air lock leakage test, and both air lock doors must be operable. The closure of a single door in an air lock will maintain primary containment operability since each door is designed to withstand the peak primary containment pressure calculated to occur following a DBA. The action provisions have been modified to allow entry and exit to perform repairs on an affected air lock component or the removal of personnel should a component failure prevent exiting in the normal manner. The ability to open the operable door, even if it means the primary containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the primary containment during the short time in which the operable door is expected to be open. The operable door must be immediately closed after each entry and exit.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water

3.7 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1000 psig.

Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber design pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber. (Ref. Section 5.2.3 FSAR)

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 48 psig which is below the design of 62 psig. Maximum water volume of 11, 655 ft^3 results in a

4.7 <u>SURVEILLANCE REQUIREMENT BASES</u> (Cont'd.)

The data reduction methods of the applicable ANSI standard will be applied for the integrated leak rate tests as specified in Appendix J of 10 CFR 50.

The penetration and air purge piping leakage test frequency, along with the containment leak rate tests, is adequate to allow detection of leakage trends. Whenever a double-gasketed penetration (primary containment head equipment hatches and the suppression chamber access hatch) is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. The test pressure of 48 psig is consistent with the accident analyses and the maximum preoperational leak rate test pressure. It is expected that the majority of the leakage from valves, penetrations and seals would be into the reactor building. However, it is possible that leakage into other parts of the facility could occur. Such leakage paths that may affect significantly the consequences of accidents are to be minimized.

Maintaining primary containment air locks operable requires compliance with the leakage-rate test requirements of 10 CFR 50, Appendix J. The periodic testing requirements verify that the air lock leakage does not exceed the specified allowed fraction of the overall primary containment leakage rate. The frequencies are required by 10 CFR 50, Appendix J. Periodic testing of the interlock mechanism demonstrates that the interlock will function as designed and that simultaneous inner and outer door opening will not inadvertently occur.

The results of the loss-of-coolant accident analyses presented in Amendment No. 18 of the SAR indicates that fission products would not be released directly to the environs because of leakage from the main steam line isolation valves due to holdup in the steam system complex. Although this effect would indicate that an adequate margin exists with regard to the release of fission products, a program will be undertaken to further reduce the potential for such leakage to bypass the standby gas treatment system.

Monitoring the nitrogen makeup requirements of the inerting system provides a method of observing leak rate trends and would detect gross leaks in a very short time. This equipment must be periodically removed from service for test and maintenance, but this out-of-service time will be kept to a practical minimum.

Surveillance of the reactor building-pressure suppression chamber vacuum breakers consists of operability checks and leakage tests

4.7 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

(conducted as part of the containment leak-tightness test). These vacuum breakers are normally in the closed position and open only during tests or a post accident condition. As a result, a testing frequency of 3 months for operability is considered justified for this equipment. Inspections and calibrations are performed during refueling outages, this frequency being based on experience and judgement.

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-249

DRESDEN NUCLEAR POWER STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 119
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 June 1, 1992, 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. 119 , 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 within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

James E. Dyer

James E. Dyer, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 11, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 119

FACILITY OPERATING LICENSE NO. DPR-25

DOCKET NO. 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 and contain marginal lines indicating the area of change. Pages indicated with an asterisk were inadvertently deleted in Amendment No. 117.

REMOVE	INSERT
iii	iii
v	v
1.0-3	1.0-3
3/4.7-7	3/4.7-7
3/4.7-9	3/4.7-9
3/4.7-19	3/4.7-19
-	3/4.7-19a
-	3/4.7-19b
-	*3/4.7-31
-	*3/4.7-32
B 3/4.7-33	B 3/4.7-33
B 3/4.7-34	B 3/4.7-34
B 3/4.7-43	B 3/4.7-43
-	B 3/4.7-43a

(Table of Contents, Cont'd.)

(Table of Contents, Cont'd.)

1.0 DEFINITIONS (Cont'd)

R. Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All manual containment isolation valves on lines connecting to the reactor coolant system or containment which are not required to be open during accident conditions are closed, or comply with the requirements of Specification 3.7.D.

2. Each primary containment air lock is in compliance with the requirements of Specification 3.7.A.8.

3. All automatic containment isolation valves are operable or deactivated in the isolated position, or comply with the requirements of Specification 3.7.D.

4. All blind flanges and manways are closed.

S. Protective Instrumentation Definitions

1. Instrument Channel - An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel.

2. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

3. Protective Action - An action initiated by the protection system when a limit is reached. A protective action can be at a channel or system level.

4. Protective Function - A system protective action which results from the protective action of the channels monitoring a particular plant condition.

T. Rated Neutron Flux - Rated neutron flux is the neutron flux that corresponds to a steady-state power level of 2527 thermal megawatts.

U. Rated Thermal Power - Rated thermal power means a steady-state power level of 2527 thermal megawatts.

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

(b) Deleted

(c) 11.5 SCF per hour for any main steam isolation valve at a test pressure of 25 psig.

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

c. If two consecutive Type A tests fail to meet either 75 percent of L_a or 75 percent of L_t, a Type A test shall be performed at each shutdown for refueling or approximately every 18 months until two consecutive Type A tests meet the above requirements, at which time the normal test schedule may be resumed.

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

(1) Confirms the accuracy of the test by verifying that the difference

3.7 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.7 SURVEILLANCE REQUIREMENTS (Cont'd.)

(2) Bolted double-gasketed seals which shall be tested at a pressure of 48 psig whenever the seal is closed after being opened and each operating cycle.

(3) Air locks shall be tested per Specification 4.7.A.8.

(4) Deleted

f. Continuous Leak Rate Monitor

(1) When the primary containment is inerted, the containment shall be continuously monitored for gross leakage by review of the inerting system make-up requirements.

3.7 LIMITING CONDITION FOR OPERATION (cont'd)

shall be initiated and the reactor shall be in a cold shutdown condition in the following 24 hours.

8. Primary Containment Air Locks

a. 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 airlock leakage rate of less than or equal to 0.05 La at Pa, 48 psig.

b. With one primary containment air lock door inoperable:

(1) Maintain at least the operable air lock door closed[a] and either restore the inoperable air lock door to operable status within 24 hours or lock the operable air lock door closed.

4.7 SURVEILLANCE REQUIREMENTS (cont'd)

8. Primary Containment Air Locks

a. Each primary containment air lock shall be demonstrated operable:

(1) By conducting an overall air lock leakage test at Pa, 48 psig and verifying that the overall air lock leakage rate is within its limit:

(a) Within 72 hours of air lock opening when 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[b], and

(c) Prior to establishing Primary Containment Integrity following air lock opening.

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a 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.

b The provisions of Specification 1.0.CC are not applicable.

3.7 LIMITING CONDITION FOR OPERATION (cont'd)

(2) 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[a] at least once per 31 days.

(3) Otherwise, be in at least hot shutdown within the next 12 hours and in cold shutdown within the following 24 hours.

c. With the primary containment air interlock mechanism inoperable:

(1) Operations may continue provided the air lock is otherwise operable and entry and exit of the primary containment is administratively controlled by a dedicated individual.

(2) Otherwise, restore the air lock interlock mechanism to operable status within 24 hours or lock the operable air lock door closed and verify that the operable air lock door is locked closed at least once per 31 days.

4.7 SURVEILLANCE REQUIREMENTS (cont'd)

(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 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.

3.7 LIMITING CONDITION FOR OPERATION (cont'd)

d. With the primary containment air lock inoperable, except as a result of an inoperable air lock door or air lock interlock mechanism:

(1) Maintain at least one air lock door closed.

(2) 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 at least cold shutdown within the following 24 hours.

B. Standby Gas Treatment System

1. Two separate and independent standby gas treatment system subsystems shall be operable at all times when secondary containment integrity is required, except as specified in sections 3.7.B.1(a) and (b).

a. After one of the standby gas treatment system subsystems is made or found to be inoperable for any reason, reactor operation and fuel handling is permissible only during the succeeding seven days, provided that all active components in the other. standby gas treatment subsystem shall be operable.

4.7 SURVEILLANCE REQUIREMENTS (cont'd)

B. Standby Gas Treatment System

1. At least once per month, initiate from the control room 4000 cfm (plus or minus 10%) flow through each subsystem of standby gas treatment system for at least 10 hours with the subsystem heaters operating at rated power.

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3.7 LIMITING CONDITION FOR OPERATION BASES

A. Primary Containment - The integrity of the primary containment and operation of the emergency core cooling system in combination, limit the off-site doses to values less than those suggested in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such a violation exists whenever the reactor is critical and above atmospheric pressure. An exception is made to this requirement during initial core loading and while the low power test program is being conducted during initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required. There will be no pressure on the system at this time which will greatly reduce the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedures will be in effect again to minimize the probability of an accident occurring. Procedures and the Rod Worth Minimizer would limit control worth to preclude a peak fuel enthalpy of 280 cal/gm. In addition, in the unlikely event that an excursion did occur, the reactor building and standby gas treatment system, which shall be operational during this time, offers a significant barrier to keep off-site doses well within 10 CFR 100.

The primary containment air lock's structural integrity and leak tightness are essential to the successful mitigation of a design basis accident event (DBA). The air lock is required to be operable whenever primary containment integrity is required. For the air lock to be considered operable, the air lock interlock mechanism must be operable, the air lock must be in compliance with the 10 CFR 50, Appendix J, Type B air lock leakage test, and both air lock doors must be operable. The closure of a single door in an air lock will maintain primary containment operability since each door is designed to withstand the peak primary containment pressure calculated to occur following a DBA. The action provisions have been modified to allow entry and exit to perform repairs on an affected air lock component or the removal of personnel should a component failure prevent exiting in the normal manner. The ability to open the operable door, even if it means the primary containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the primary containment during the short time in which the operable door is expected to be open. The operable door must be immediately closed after each entry and exit.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water

3.7 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1000 psig.

Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss of coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 62 psig, the suppression chamber design pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber. (Ref. Section 5.2.3 FSAR)

Using the minimum or maximum water volumes given in the specification, containment pressure during the design basis accident is approximately 48 psig which is below the design of 62 psig. Maximum water volume of 11, 655 ft^3 results in a

4.7 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

The data reduction methods of the applicable ANSI standard will be applied for the integrated leak rate tests as specified in Appendix J of 10 CFR 50.

The penetration and air purge piping leakage test frequency, along with the containment leak rate tests, is adequate to allow detection of leakage trends. Whenever a double-gasketed penetration (primary containment head equipment hatches and the suppression chamber access hatch) is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. The test pressure of 48 psig is consistent with the accident analyses and the maximum preoperational leak rate test pressure. It is expected that the majority of the leakage from valves, penetrations and seals would be into the reactor building. However, it is possible that leakage into other parts of the facility could occur. Such leakage paths that may affect significantly the consequences of accidents are to be minimized.

Maintaining primary containment air locks operable requires compliance with the leakage-rate test requirements of 10 CFR 50, Appendix J. The periodic testing requirements verify that the air lock leakage does not exceed the specified allowed fraction of the overall primary containment leakage rate. The frequencies are required by 10 CFR 50, Appendix J. Periodic testing of the interlock mechanism demonstrates that the interlock will function as designed and that simultaneous inner and outer door opening will not inadvertently occur.

The results of the loss-of-coolant accident analyses presented in Amendment No. 18 of the SAR indicates that fission products would not be released directly to the environs because of leakage from the main steam line isolation valves due to holdup in the steam system complex. Although this effect would indicate that an adequate margin exists with regard to the release of fission products, a program will be undertaken to further reduce the potential for such leakage to bypass the standby gas treatment system.

Monitoring the nitrogen makeup requirements of the inerting system provides a method of observing leak rate trends and would detect gross leaks in a very short time. This equipment must be periodically removed from service for test and maintenance, but this out-of-service time will be kept to a practical minimum.

Surveillance of the reactor building-pressure suppression chamber vacuum breakers consists of operability checks and leakage tests

4.7 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

(conducted as part of the containment leak-tightness test). These vacuum breakers are normally in the closed position and open only during tests or a post accident condition. As a result, a testing frequency of 3 months for operability is considered justified for this equipment. Inspections and calibrations are performed during refueling outages, this frequency being based on experience and judgement.

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

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-254

QUAD CITIES NUCLEAR POWER STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 145
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 June 1, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 145 , 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 within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

James E. Dyer

James E. Dyer, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 11, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 145

FACILITY OPERATING LICENSE NO. DPR-29

DOCKET NO. 50-254

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 and contain marginal lines indicating the area of change.

REMOVE	INSERT
1.0-3	1.0-3
3.7/4.7-5	3.7/4.7-5
-	3.7/4.7-11a
-	3.7/4.7-11b
-	3.7/4.7-11c
-	3.7/4.7-11d
3.7/4.7-12	3.7/4.7-12
3.7/4.7-21	3.7/4.7-21
-	3.7/4.7-21a
3.7/4.7-28	3.7/4.7-28

M. Operable - A system, subsystem, train, component, or device shall be operable when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem train, component or device to perform its function(s) are also capable of performing their related support function(s).

N. Operating - Operating means that a system, subsystem, train, component or device is performing its intended functions in its required manner.

O. Operating Cycle - Interval between the end of one refueling outage for a particular unit and the end of the next subsequent refueling outage for the same unit.

P. Primary Containment Integrity - Primary containment integrity means that the drywell and pressure suppression chamber are intact and all of the following conditions are satisfied:

1. All manual containment isolation valves on lines connecting to the reactor coolant system or containment which are not required to be open during accident conditions are closed.

2. Each primary containment air lock is in compliance with the requirements of Specification 3.7.A.7.

3. All automatic containment isolation valves are operable or deactivated in the isolation position.

4. All blind flanges and manways are closed.

Q. Protective Instrumentation Definitions

1. Channel - A channel is an arrangement of a sensor and associated components used to evaluate plant variables and produce discrete outputs used in logic. A channel terminates and loses its identity where individual channel outputs are combine in a logic.

2. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

3) Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total measured leakage at P_a, 48 psig, or P_t, 25 psig.

d. Deleted

d. Type B and C tests shall be conducted at P_a, 48 psig, at intervals no greater than 24 months except for tests involving:

1) Air locks shall be tested per Specification 4.7.A.7.

2) Main steam isolation valves which shall be leak tested at least once per operating cycle, of a frequency not to exceed 24 months, at a pressure of 25 psig.

3) Bolted double-gasketed seals which shall be tested at a pressure of 48 psig whenever the seal is closed after being opened and each operating cycle.

4) While valve MO1-200-1 is inoperable, valves MO1-220-2, MO1-220-3, and MO1-220-4 shall be VERIFIED closed after each closure.

5) The pathways identified in Table 4.7-1, which will not be tested until the end of cycle 11 refueling outage.

c. This pressure differential may be decreased to less than 1.20 PSID for a maximum of 4 hours during required operability testing of the HPCI system pump, the RCIC system pump, the drywell-pressure suppression chamber vacuum breakers, and reactor pressure relief valves.

d. If the Specifications of 3.7.A.6.c cannot be met, and the differential pressure cannot be restored within the subsequent six (6) hours period, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition in the following 24 hours.

7. Primary Containment Air Locks

a. 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 less than or equal to 0.05 L_a at P_a, 48 psig.

b. With one primary containment air lock door inoperable:

(1) Maintain at least the Operable air lock door closed[a] and either restore the inoperable air lock door to Operable status within 24 hours or lock the Operable air lock door closed.

7. Primary Containment Air Locks

a. Each primary containment air lock shall be demonstrated Operable:

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

(a) Within 72 hours of air lock opening when 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[b], and

(c) Prior to establishing Primary Containment Integrity following air lock opening.

---

[a] 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.

[b] The provisions of Specification 1.0.DD are not applicable.

(2) 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* at least once per 31 days.

(3) Otherwise, be in at least hot shutdown within the next 12 hours and in cold shutdown within the following 24 hours.

c. With the primary containment air lock interlock mechanism inoperable:

(1) Operations may continue provided the air lock is otherwise Operable and entry and exit of the primary containment is administratively controlled by a dedicated individual.

(2) Otherwise, restore the air lock interlock mechanism to Operable status within 24 hours or lock the Operable air lock door closed and verify that the Operable air lock door is locked closed at least once per 31 days.

d. With the primary containment air lock inoperable, except as a result of an inoperable air lock door or air lock interlock mechanism:

(1) Maintain at least one air lock door closed.

(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.

---

* 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.

(2) 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 at least cold shutdown within the following 24 hours.

B. Standby Gas Treatment System

1. Two separate and independent standby gas treatment circuits shall be operable at all times when secondary containment integrity is required, except as specified in sections 3.7.B.1.(a) and (b).

   a. After one of the standby gas treatment circuits is made or found to be inoperable for any reason, reactor operation and fuel handling is permissible only during the succeeding 7 days, provided that all active components in the other standby gas treatment system shall be demonstrated to be operable within 2 hours and daily thereafter. Within 36 hours following the 7 days, the reactor shall be placed in a condition for which the standby gas treatment system is not required in accordance with Specification 3.7.C.1(a) through (d).

B. Standby Gas Treatment System

1. At least once per month, initiate from the control room 4000 cfm (± 10%) flow through both circuits of the standby gas treatment system for at least 10 hours with the circuit heaters operating at rated power.

   a. Within 2 hours from the time that one standby gas treatment system circuit is made or found to be inoperable for any reason and daily thereafter for the next succeeding 7 days, initiate from the control room 4000 cfm (± 10%) flow through the operable circuit of the standby treatment system for at least 10 hours with the circuit heaters operating.

3.7 LIMITING CONDITIONS FOR OPERATION BASES

A. Primary Containment

The integrity of the primary containment and operation of the emergency core cooling system, in combination, limit the offsite doses to values less than those suggested in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such violation exists whenever the reactor is critical and above atmospheric pressure. An exception is made to this requirement during initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required. There will be no pressure on the system at this time, which will greatly reduce the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedure will be in effect again to minimize the probability of an accident occurring. Procedures and the rod worth minimizer would limit control rod worth to preclude a peak fuel enthalpy of 280 cal/gm. In addition, in the unlikely event that an excursion did occur, the reactor building standby gas treatment system, which will be operational during this time, offers a sufficient barrier to keep offsite doses well within 10 CFR 100 guidelines.

The primary containment air lock's structural integrity and leak tightness are essential to the successful mitigation of a design basis accident event (DBA). The air lock is required to be operable whenever primary containment integrity is required. For the air lock to be considered operable, the air lock interlock mechanism must be operable, the air lock must be in compliance with the 10 CFR 50, Appendix J, Type B air lock leakage test, and both air lock doors must be operable. The closure of a single door in an air lock will maintain primary containment operability since each door is designed to withstand the peak primary containment pressure calculated to occur following a DBA. The action provisions have been modified to allow entry and exit to perform repairs on an affected air lock component or the removal of personnel should a component failure prevent exiting in the normal manner. The ability to open the operable door, even if it means the primary containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the primary containment during the short time in which the operable door is expected to be open. The operable door must be immediately closed after each entry and exit.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1000 psig.

Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss-of-coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 56 psig, the suppression chamber design pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber. (Ref. Section 6.2.1 FSAR)

Using the minimum or maximum water volume given in the specification, containment pressure during the design-basis accident is approximately 48 psig, which is below the design value of 56 psig. Maximum water volume of 115,655 ft^3 results in a downcomer submergence of 4 feet; the minimum volume of 112,200 ft^3 results in a submergence approximately 4 inches less. The majority of the Bodega tests (Reference 1) were run with a submerged length of 4 feet and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate.

These doses are also based on the assumption of no holdup in the secondary containment resulting in direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide margin between expected offsite doses and 10 CFR 100 guidelines.

Although the dose calculations suggest that the accident leak rate could be allowed to increase to about 2.6%/day before the guideline thyroid dose value given in 10 CFR 100 would be exceeded, establishing the test limit of 1.0%/day provides an adequate margin of safety to assure the health and safety of the general public. It is further considered that the allowable leak rate should not deviate significantly from the containment design value to take advantage of the design leak tightness capability of the structure over its service lifetime. Additional margin to maintain the containment in the as-built condition is achieved by establishing the allowable operational leak rate. The allowable operational leak rate is derived by multiplying the maximum allowable leak rate by 0.75, thereby providing a 25% margin to allow for leakage deterioration which may occur during the period between leak rate tests.

The primary containment leak rate test frequency is based on maintaining adequate assurance that the leak rate remains within the specification. Allowing the test intervals to be extended up to 8 months permits some flexibility needed to have the tests coincide with scheduled or unscheduled shutdown periods.

The data reduction methods of ANSI N45.4-1972 will be applied for integrated leak rate tests.

The penetration and air purge piping leakage test frequency, along with the containment leak rate tests, is adequate to allow detection of leakage trends. Whenever a double-gasketed penetration (primary containment head equipment hatches and the suppression chamber access hatch) is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. The test pressure of 48 psig is consistent with the accident analyses and the maximum preoperational leak rate test pressure. It is expected that the majority of the leakage from valves, penetrations, and seals would be into the reactor building. However, it is possible that leakage into other parts of the facility could occur. Such leakage paths that may affect significantly the consequences of accidents are to be minimized.

Maintaining primary containment air locks operable requires compliance with the leakage-rate test requirements of 10 CFR 50, Appendix J. The periodic testing requirements verify that the air lock leakage does not exceed the specified allowed fraction of the overall primary containment leakage rate. The frequencies are required by 10 CFR 50, Appendix J. Periodic testing of the interlock mechanism demonstrates that the interlock will function as designed and that simultaneous inner and outer door opening will not inadvertently occur.

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

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-265

QUAD CITIES NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 141
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 June 1, 1992, 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. 141 , 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 within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

James E. Dyer

James E. Dyer, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 11, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 141

FACILITY OPERATING LICENSE NO. DPR-30

DOCKET NO. 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 and contain marginal lines indicating the area of change.

REMOVE	INSERT
1.0-3	1.0-3
3.7/4.7-3	3.7/4.7-3
-	3.7/4.7-6b
-	3.7/4.7-6c
-	3.7/4.7-6d
3.7/4.7-7	3.7/4.7-7
3.7/4.7-11	3.7/4.7-11
-	3.7/4.7-11a
3.7/4.7-16	3.7/4.7-16

2. Each primary containment air lock is in compliance with the requirements of Specification 3.7.A.7.

3. All automatic containment isolation valves are operable or deactivated in the isolation position.

4. All blind flanges and manways are closed.

Q. **Protective Instrumentation Definitions**

1. Channel - A channel is an arrangement of a sensor and associated components used to evaluate plant variables and produce discrete outputs used in logic. A channel terminates and loses its identity where individual channel outputs are combine in a logic.

2. Trip System - A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system or the coincident tripping of two trip systems.

3. Protective Action - An action initiated by the protection system when a limit is reached. A protective action can be a the channel or system level.

4. Protective Function - A system protective action which results from the protective action of the channels monitoring a particular plant condition.

R. **Rated Neutron Flux** - Rated neutron flux is the flux that corresponds to a stead-state power level of 2511 thermal megawatts.

S. **Rated Thermal Power** - Rated thermal power means a steady-state power level of 2511 thermal megawatts.

T. **Reactor Power Operation** - Reactor power operation is any operation with the mode switch in the Startup/Hot Standby or Run position with the reactor critical and above 1% rated thermal power.

U. **Reactor Vessel Pressure** - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those measured by the reactor vessel steam space detector.

V. **Refueling Outage** - Refueling outage is the period of time between the shutdown of the unit prior to a refueling and the startup of the plant subsequent to that refueling. For the purpose of designating frequency of testing surveillance, a refueling outage shall mean a regularly scheduled refueling outage; however, where such outages occur within 8 months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage.

W. **Safety Limit** - The safety limits are limits below which the reasonable maintenance of the cladding and primary system are assured. Exceeding such a limit is cause for unit shutdown, and review by the NRC before resumption of the unit operation. Operation beyond such a limit may not in itself result in serious consequences, but it indicates an operational deficiency subject to regulatory review.

X. **Secondary Containment Integrity** - Secondary containment integrity means that the reactor building is intact and the following conditions are met:

1. At least one door in each access opening is closed.

2. The standby gas treatment system is operable.

3. All reactor building automatic ventilation system isolation valves are operable or are secured in the isolated position.

b) $\leq L_t$, 1.0 percent by weight of the containment air per 24 hours at a reduced pressure of P_t, 25 psig.

2) A combined leakage rate of $\leq 0.60\ L_a$ for all penetrations and valves, except for main steam isolation valves subject to Type B and C tests when pressurized to P_a.

3) 11.5 scf per hour for any one main steam isolation valve when tested at 25 psig.

b. With the measured overall integrated containment leakage rate exceeding $0.75\ L_a$ or $0.75\ L_t$, as applicable, restore the overall integrated leakage rate(s) to $\leq 0.75\ L_a$ or $\leq 0.75\ L_t$, as applicable.

c. With the measured combined leakage rate for all penetrations and valves, except for main steam isolation valves, subject to Type B and C tests exceeding $0.60\ L_a$, restore the combine leakage rate for all penetrations and valves, except for main steam isolation valves, subject to Type B and C tests to $0.60\ L_a$.

d. Deleted

b. If any periodic Type A test fails to meet either $0.75\ L_a$ or $0.75\ L_t$, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet either $0.75\ L_a$ or $0.75\ L_t$, a Type A test shall be performed at each shutdown for refueling or approximately every 18 months, whichever occurs first, until two consecutive Type A tests meet either $0.75\ L_a$ or $0.75\ L_t$, at which time the above test schedule may be resumed.

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

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

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

3) Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total measured leakage at P_a, 48 psig, or P_t, 25 psig.

d. Type B and C tests shall be conducted at P_a, 48 psig, at intervals no greater than 24 months except for tests involving:

1) Air locks shall be tested per Specification 4.7.A.7.

c. This pressure differential may be decreased to less than 1.20 PSID for a maximum of 4 hours during required operability testing of the HPCI system pump, the RCIC system pump, the drywell-pressure suppression chamber vacuum breakers, and reactor pressure relief valves.

d. If the Specifications of 3.7.A.6.c cannot be met, and the differential pressure cannot be restored within the subsequent six (6) hours period, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition in the following 24 hours.

7. Primary Containment Air Locks

a. 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 less than or equal to 0.05 L_a at P_a, 48 psig.

b. With one primary containment air lock door inoperable:

(1) Maintain at least the Operable air lock door closed[a] and either restore the inoperable air lock door to Operable status within 24 hours or lock the Operable air lock door closed.

7. Primary Containment Air Locks

a. Each primary containment air lock shall be demonstrated Operable:

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

(a) Within 72 hours of air lock opening when 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[b], and

(c) Prior to establishing Primary Containment Integrity following air lock opening.

---

[a] 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.

[b] The provisions of Specification 1.0.DD are not applicable.

(2) 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* at least once per 31 days.

(3) Otherwise, be in at least hot shutdown within the next 12 hours and in cold shutdown within the following 24 hours.

c. With the primary containment air lock interlock mechanism inoperable:

(1) Operations may continue provided the air lock is otherwise Operable and entry and exit of the primary containment is administratively controlled by a dedicated individual.

(2) Otherwise, restore the air lock interlock mechanism to Operable status within 24 hours or lock the Operable air lock door closed and verify that the Operable air lock door is locked closed at least once per 31 days.

d. With the primary containment air lock inoperable, except as a result of an inoperable air lock door or air lock interlock mechanism:

(1) Maintain at least one air lock door closed.

(2) 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 at least cold shutdown within the following 24 hours.

(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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* 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.

B. Standby Gas Treatment System

1. Two separate and independent standby gas treatment circuits shall be operable at all times when secondary containment integrity is required, except as specified in sections 3.7.B.1.(a) and (b).

   a. After one of the standby gas treatment circuits is made or found to be inoperable for any reason, reactor operation and fuel handling is permissible only during the succeeding 7 days, provided that all active components in the other standby gas treatment system shall be demonstrated to be operable within 2 hours and daily thereafter. Within 36 hours following the 7 days, the reactor shall be placed in a condition for which the standby gas treatment system is not required in accordance with Specification 3.7.C.1(a) through (d).

   b. If both standby gas treatment system circuits are not operable, within 36 hours the reactor shall be placed in a condition for which the standby gas treatment system is not required in accordance with Specification 3.7.C.1.(a) through (d).

B. Standby Gas Treatment System

1. At least once per month, initiate from the control room 4000 cfm (± 10%) flow through both circuits of the standby gas treatment system for at least 10 hours with the circuit heaters operating at rated power.

   a. Within 2 hours from the time that one standby gas treatment system circuit is made or found to be inoperable for any reason and daily thereafter for the next succeeding 7 days, initiate from the control room 4000 cfm (± 10%) flow through the operable circuit of the standby treatment system for at least 10 hours with the circuit heaters operating.

3.7 LIMITING CONDITIONS FOR OPERATION BASES

A. Primary Containment

The integrity of the primary containment and operation of the emergency core cooling system, in combination, limit the offsite doses to values less than those suggested in 10 CFR 100 in the event of a break in the primary system piping. Thus, containment integrity is specified whenever the potential for violation of the primary reactor system integrity exists. Concern about such violation exists whenever the reactor is critical and above atmospheric pressure. An exception is made to this requirement during initial core loading and while the low power test program is being conducted and ready access to the reactor vessel is required. There will be no pressure on the system at this time, which will greatly reduce the chances of a pipe break. The reactor may be taken critical during this period; however, restrictive operating procedure will be in effect again to minimize the probability of an accident occurring. Procedures and the rod worth minimizer would limit control rod worth to preclude a peak fuel enthalpy of 280 cal/gm. In addition, in the unlikely event that an excursion did occur, the reactor building standby gas treatment system, which will be operational during this time, offers a sufficient barrier to keep offsite doses well within 10 CFR 100 guidelines.

The primary containment air lock's structural integrity and leak tightness are essential to the successful mitigation of a design basis accident event (DBA). The air lock is required to be operable whenever primary containment integrity is required. For the air lock to be considered operable, the air lock interlock mechanism must be operable, the air lock must be in compliance with the 10 CFR 50, Appendix J, Type B air lock leakage test, and both air lock doors must be operable. The closure of a single door in an air lock will maintain primary containment operability since each door is designed to withstand the peak primary containment pressure calculated to occur following a DBA. The action provisions have been modified to allow entry and exit to perform repairs on an affected air lock component or the removal of personnel should a component failure prevent exiting in the normal manner. The ability to open the operable door, even if it means the primary containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the primary containment during the short time in which the operable door is expected to be open. The operable door must be immediately closed after each entry and exit.

The pressure suppression pool water provides the heat sink for the reactor primary system energy release following a postulated rupture of the system. The pressure suppression chamber water volume must absorb the associated decay and structural sensible heat released during primary system blowdown from 1000 psig.

Since all of the gases in the drywell are purged into the pressure suppression chamber air space during a loss-of-coolant accident, the pressure resulting from isothermal compression plus the vapor pressure of the liquid must not exceed 56 psig, the suppression chamber design pressure. The design volume of the suppression chamber (water and air) was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber. (Ref. Section 6.2.1 FSAR)

Using the minimum or maximum water volume given in the specification, containment pressure during the design-basis accident is approximately 48 psig, which is below the design value of 56 psig. Maximum water volume of 115,655 ft^3 results in a downcomer submergence of 4 feet; the minimum volume of 112,200 ft^3 results in a submergence approximately 4 inches less. The majority of the Bodega tests (Reference 1) were run with a submerged length of 4 feet and with complete condensation. Thus, with respect to downcomer submergence, this specification is adequate.

Experimental data indicate that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. 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 relief valve inadvertently opens or sticks open. This action would include: (1) use of all available means to close the valve, (2) initiation of suppression pool water cooling heat exchangers, (3) initiation of reactor shutdown, and (4) if other relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open relief valve to assure mixing and uniformity of energy insertion to the pool.

The maximum temperature at the end of the blowdown tested during the Humboldt Bay (Reference 2) and Bodega Bay tests was 170°F; this is conservatively taken to be the limit for complete condensation of the reactor coolant, although condensation would occur for temperatures above 170°F.

multiplying the maximum allowable leak rate by 0.75, thereby providing a 25% margin to allow for leakage deterioration which may occur during the period between leak rate tests.

The primary containment leak rate test frequency is based on maintaining adequate assurance that the leak rate remains within the specification. Allowing the test intervals to be extended up to 8 months permits some flexibility needed to have the tests coincide with scheduled or unscheduled shutdown periods.

The data reduction methods of ANSI N45.4-1972 will be applied for integrated leak rate tests.

The penetration and air purge piping leakage test frequency, along with the containment leak rate tests, is adequate to allow detection of leakage trends. Whenever a double-gasketed penetration (primary containment head equipment hatches and the suppression chamber access hatch) is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. The test pressure of 48 psig is consistent with the accident analysis and the maximum preoperational leak rate test pressure. It is expected that the majority of the leakage from valves, penetrations, and seals would be into the reactor building. However, it is possible that leakage into other parts of the facility could occur. Such leakage paths that may affect significantly the consequences of accidents are to be minimized.

Maintaining primary containment air locks operable requires compliance with the leakage-rate test requirements of 10 CFR 50, Appendix J. The periodic testing requirements verify that the air lock leakage does not exceed the specified allowed fraction of the overall primary containment leakage rate. The frequencies are required by 10 CFR 50, Appendix J. Periodic testing of the interlock mechanism demonstrates that the interlock will function as designed and that simultaneous inner and outer door opening will not inadvertently occur.

The results of the loss-of-coolant accident analysis referenced in Section 6.2.4.1 of the FSAR indicate that fission products would not be released directly to the environs because of leakage from the main steam line isolation valves due to holdup on the steam system complex. Although this effect would indicate that an adequate margin exists with regard to the release of fission products, a program will be undertaken to further reduce the potential for such leakage to bypass the standby gas treatment system.

Surveillance of the reactor building-pressure suppression chamber vacuum breakers consists of operability checks and leakage tests (conducted as part of the containment leak tightness tests). These vacuum breakers are normally in the closed position and open only during tests or an accident condition. As a result, a testing frequency of 3 months for operability is considered justified for this equipment. Inspections and calibrations are performed during refueling outages, this frequency being based on experience and judgment.

Pressure suppression chamber-drywell vacuum breakers monthly operability tests are performed to check the capability of the disks to open and close and to verify that the position indication and alarm circuits function properly. The disks must open during accident conditions and during transient additions of energy through relief valves. This periodic operation of the disks and the quality of equipment justify the frequency of operability tests of this equipment.

Following each quarterly operability test, a differential pressure decay rate test is performed to verify that leakage from the drywell to the suppression chamber is within specified limits.

Measurement of force to open, calibration of position switches, inspection of equipment, and functional testing are performed during each refueling outage. This frequency is based on equipment quality, experience, and judgement. Also, a more stringent differential pressure decay rate test is performed during refueling outages than is performed monthly. This test is performed to verify that total leakage paths between the drywell and suppression chamber are not in excess of the equivalent to a 1-inch orifice.

This small leakage path is only a small fraction of the allowable, thus integrity of the containment system is assured prior to startup following each refueling outage (Reference 1).