Distribution
Docket
ORB #3
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PO'Connor
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DEisenhut
ACRS (16)
OPA (CMiles)
DRoss
TBAbernathy
JRBuchanan
RDiggs

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

JUNE 7 1978

Commonwealth Edison Company
ATTN: Mr. Cordell Reed
Assistant Vice President
P. O. Box 767
Chicago, Illinois 60690

Gentlemen:

In response to your request for license amendments dated November 30, 1976 and supplements thereto dated April 18 and April 19, 1977, the Commission has issued the enclosed Amendment Nos. 37, 35, 46 and 46 to Facility Operating License Nos. DPR-19, DPR-25, DPR-29, and DPR-30 for the Dresden Station Units Nos. 2 and 3 and Quad-Cities Station Units Nos. 1 and 2, respectively.

These amendments incorporate provisions into the Technical Specifications for the facilities which establish limiting conditions for operation and surveillance requirements for drywell to suppression chamber differential pressure control and suppression pool water level.

These requirements provide assurance that operation will be in accordance with the assumptions utilized in the plant-unique analyses for the facilities, which were performed in conjunction with the Mark I Containment Short Term Program evaluation.

The enclosed license amendments reflect those changes to your original request for license amendments which have been agreed to in discussions with your staff. These changes have been made to provide consistent requirements for all Mark I containment facilities.

Copies of the related Safety Evaluation and Notice of issuance are also enclosed.

Sincerely,

Original signed by

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

CONST 1

Enclosures and ccs:
See next page

OFFICE	DOR	DOR	DOR	ORB #2	ORB #3
SURNAME	Sheppard/Smith	OConnor/Bevan	CGrimes	DZiemann	GLear
DATE	5/ /78	5/ /78	5/ /78	5/ /78	5/ /78

NRC FORM 318 (9-76) NRCM 0240 ☆ U.S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

Enclosures:
1. Amendment No. 37 to License No. DPR-19
2. Amendment No. 35 to License No. DPR-25
3. Amendment No. 46 to License No. DPR-29
4. Amendment No. 46 to License No. DPR-30
5. Safety Evaluation
6. Notice

cc w/enclosures: See next page

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Plant Superintendent
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Morris, Illinois 60450

Anthony Z. Roisman
Natural Resources Defense Council
917 15th Street, N. W.
Washington, D. C. 20005

Morris Public Library
604 Liberty Street
Morris, Illinois 60451

Illinois Department of Public Health
ATTN: Chief, Division of Nuclear Safety
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Springfield, Illinois 62761

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Chairman, Board of Supervisors of Grundy County
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN STATION UNIT NO. 2

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 37
License No. DPR-19

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated November 30, 1976, as supplemented by letters dated April 18 and April 19, 1977, 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 Provisional Operating License No. DPR-19 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 37, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 7, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 37

PROVISIONAL OPERATING LICENSE NO. DPR-19

DOCKET NO. 50-237

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
44	44
45	45
108A	108A
117b	117b
127	127

Add page 127a

TABLE 4.2.1 (cont)

Instrument Channel	Instrument Functional Test (2)	Calibration (2)	Instrument Check (2)
ISOLATION CONDENSER ISOLATION			
1. Steam Line High Flow	(1)	Once/3 Months	None
2. Condensate Line High Flow	(1)	Once/3 Months	None
HPCI ISOLATION			
1. Steam Line High Flow	(1)	Once/3 Months	None
2. Steam Line Area High Temperature	Refueling Outage	Refueling Outgage	None
3. Low Reactor Pressure	(1)	Once/3 Months	None
REACTOR BUILDING VENTILATION SYSTEM ISOLATION AND STANDBY GAS TREATMENT SYSTEM INITIATION			
1. Ventilation Exhaust Duct Radiation Monitors	(1)	Once/3 Months	Once/Day
2. Refueling Floor Radiation Monitors	(1)	Once/3 Months	Once/Day
STEAM JET-AIR EJECTOR OFF-GAS ISOLATION			
1. Radiation Monitors	(1) (3)	Once/3 Months (4)	Once/Day
CONTAINMENT MONITORING			
1. Pressure			
a. -5 in. Hg to +5 psig indicator	None	Once/3 Months	Once/Day
b. 0 to 75 psig indicator	None	Once/3 Months	None
2. Temperature	None	Refueling Outage	Once/Day
3. Drywell-Torus differential pressure (5)(6) (0-3 psid)	None	Once/6 months (two channels operable) Once/month (one channel operable)	None
4. Torus water level (5)(6)	None	Once/6 months	
a. $\pm$25 in. wide range indicator			
b. $\overline{1}$8 in. sight glass			

TABLE 4.2.1 (cont)

Notes:

1. Initially once per month until exposure hours (M as defined on Figure 4.1.1) is 2.0×10^5; thereafter, according to Figure 4.1.1 with an interval not less than one month nor more than three months. The compilation of instrument failure rate data may include data obtained from other Boiling Water Reactors for which the same design instrument operates in an environment similar to that of Dresden Unit 3.

2. Functional test calibrations and instrument checks are not required when these instruments are not required to be operable or are tripped. Functional tests shall be performed before each startup with a required frequency not to exceed once per week. Calibrations shall be performed during each startup or during controlled shutdowns with a required frequency not to exceed once per week. Instrument checks shall be performed at least once per week. Instrument checks shall be performed at least once per day during those periods when the instruments are required to be operable.

3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel. See Note 4.

4. These instrument channels will be calibrated using simulated electrical signals once every three months. In addition, calibration including the sensors will be performed during each refueling outage.

5. A minimum of two channels is required.

6. From and after the date that one of these parameters [... either drywell-torus differential pressure or torus water level indication] is reduced to one indication, continued operation is not permissible beyond thirty days, unless such instrumentation is sooner made operable. In the event that all indications of these parameters [...either drywell-torus differential pressure or torus water level] is disabled and such indication cannot be restored in six (6) hours, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition in twenty four hours.

above. In connection with such testing, the pool temperature must be reduced to below the normal power operation limit specified in (1) above within 24 hours.

(3) The reactor shall be scrammed from any operating condition if the pool temperature reaches 110°F. Power operation shall not be resumed until the pool temperature is reduced below normal power operation limit specified in (1) above.

(4) During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 150 psig at normal cooldown rates if the pool temperature reaches 120°F.

d. Maximum downcomer submergence is 4.00 ft.

e. Minimum downcomer submergence is 3.67 ft.

2. Primary containment integrity small be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 Mw(t).

d. A visual inspection of the suppression chamber interior, including water line regions, shall be made at each major refueling outage.

2. The primary containment integrity shall be demonstrated by either Method A or Method B, as follows:

a. Integrated Primary Containment Leak Test (IPCLT)

d. Whenever the reactor is in power operation, the primary containment oxygen sampling system shall be operable. If this specification cannot be met, the system must be restored to an operable condition within 7 days or the reactor must be taken out of power operation.

e. The maximum containment repressurization pressure using the containment makeup inerting system shall be 26 psig.

7. Drywell Suppression Chamber Differential Pressure

A. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.00 psid except as specified in (1), (2), and (3) below:

(1) This differential shall be established within the 24 hour period subsequent to placing the reactor mode switch into the run mode during a startup and may be relaxed 24 hours prior to a reactor shutdown when the provisions of 3.7.A.5 (b) apply.

(2) This differential may be decreased to less than 1.00 psid for a maximum of 4 hours during required operability testing of the drywell pressure suppression chamber vacuum breakers, HPCI testing and reactor pressure relief valve testing.

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

d. The containment oxygen analyzing system shall be functionally tested once per week and shall be calibrated once per 6 months.

7. Drywell Suppression Chamber Differential Pressure

a. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift when the differential pressure is required.

A means to determined post LOCA containment oxygen concentration is necessary to readily enable the reactor operator to take appropriate action to control containment atmosphere. In the interim, prior to installation of the CAD and associated monitoring systems, the containment oxygen analyzing system will be available.

The maximum containment repressurization pressure of 26 psi provides adequate margin to containment design pressure and a delay time prior to purge which results in acceptable purge doses.

Following a LOCA, periodic operation of the drywell and torus sprays will be used to assist the natural convection and diffusion mixing of hydrogen and oxygen when other ECCS requirements are met and O_2 concentration exceeds 4%.

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed (Reference 15) 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 of 1.00 psid 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.

B. Standby Gas Treatment System and C Secondary Containment

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation, when the drywell is sealed and in service; the reactor building provides primary containment when the reactor is shutdown and the drywell is open, as during refueling. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required as well as during refueling.

(15) "Dresden Nuclear Generating Plant Units 2 & 3 Short Term Program Plant Unique Torus Support and Attached Piping Analysis", August 1976 NUTECH Report COM-01-040.

Amendment No. 37

Only one of the two standby gas treatment system circuits is needed to cleanup the reactor building atmosphere upon containment isolation. If one system is found to be inoperable, there is no immediate threat to the containment system performance. Therefore, reactor operation or refueling operation may continue while repairs are being made. If neither circuit is operable, the plant is placed in a condition that does not require a standby gas treatment system.

While only a small amount of particulates are released from the primary containment as a result of the loss of coolant accident, high-efficiency particulate filters before and after the charcoal filters are specified to minimize potential particulate release to the environment and to prevent clogging of the charcoal adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. (The in-place test results should indicate a system leak tightness of less than 1% bypass leakage for the charcoal adsorbers using halogenated hydrocarbon and a HEPA filter efficiency of at least 99% removal of DOP particulates. Laboratory carbon sample test results indicate a radioactive methyl iodide removal efficiency for expected accident conditions. Operation of the standby gas treatment circuits significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performance requirements are met as specified, the calculated doses would be less than the guidelines stated in 10 CFR 100 for the accidents analyzed).*

*Bases in parentheses will not be applicable until about December 31, 1976, when equipment modifications are completed to allow increased testing.

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the stack during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environs. One standby gas treatment fan is designed to automatically start upon containment isolation and to maintain the reactor building pressure to approximately a negative 1/4-inch water guage pressure; all leakage should be in-leakage. Should the fan fail to start, the redundant alternate fan and filter system is designed to start automatically. Each of the two fans has 200% capacity. Ref. Section 5.3.2 SAR. If one standby gas treatment system circuit is inoperable, the other circuit will be tested daily. This substantiates the availability of the operable circuit and results in no added risk; thus, reactor operation or refueling operation can continue. If neither circuit is operable the plant is brought to a condition where the system is not required.

While only a small amount of particulates are released from the pressure suppression chamber system as a result of the loss of coolant accident, high-efficiency particulate filters before and after the charcoal filters are specified to minimize potential particulate release to the environment and to prevent clogging of the charcoal adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The inplace test results should indicate a system leak tightness of less than 1% bypass leakage for the charcoal adsorbers using halogenated hydrocarbon and a HEPA filter efficiency of at least 99% removal of DOP particulates. Laboratory carbon sample test results indicate a radioactive methyl iodide removel efficiency for expected accident conditions. Operation of the standby gas treatment circuits significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performance requirements are met as specified, the calculated doses would be less than the guidelines stated in 10 CFR 100 for the accidents analyzed.

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-249

DRESDEN STATION UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 35
License No. DPR-25

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated November 30, 1976, as supplemented by letters dated April 18 and April 19, 1977, 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 Provisional 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. 35, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 7, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 35

FACILITY OPERATING LICENSE NO. DPR-25

DOCKET NO. 50-249

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
44	44
45	45
108A	108A
117b	117b
127	127

Add page 127a

TABLE 4.2.1 (cont)

Instrument Channel	Instrument Functional Test (2)	Calibration (2)	Instrument Check (2)
ISOLATION CONDENSER ISOLATION			
1. Steam Line High Flow	(1)	Once/3 Months	None
2. Condensate Line High Flow	(1)	Once/3 Months	None
HPCI ISOLATION			
1. Steam Line High Flow	(1)	Once/3 Months	None
2. Steam Line Area High Temperature	Refueling Outage	Refueling Outgage	None
3. Low Reactor Pressure	(1)	Once/3 Months	None
REACTOR BUILDING VENTILATION SYSTEM ISOLATION AND STANDBY GAS TREATMENT SYSTEM INITIATION			
1. Ventilation Exhaust Duct Radiation Monitors	(1)	Once/3 Months	Once/Day
2. Refueling Floor Radiation Monitors	(1)	Once/3 Months	Once/Day
STEAM JET-AIR EJECTOR OFF-GAS ISOLATION			
1. Radiation Monitors	(1) (3)	Once/3 Months (4)	Once/Day
CONTAINMENT MONITORING			
1. Pressure			
a. -5 in. Hg to +5 psig indicator	None	Once/3 Months	Once/Day
b. 0 to 75 psig indicator	None	Once/3 Months	None
2. Temperature	None	Refueling Outage	Once/Day
3. Drywell-Torus differential pressure (5)(6) (0-3 psid)	None	Once/6 months (two channels operable) Once/month (one channel operable)	None
4. Torus water level (5)(6)	None	Once/6 months	
a. ± 25 in. wide range indicator			
b. 18 in. sight glass			

Amendment No. 35

TABLE 4.2.1 (cont)

Notes:

1. Initially once per month until exposure hours (M as defined on Figure 4.1.1) is 2.0×10^5; thereafter, according to Figure 4.1.1 with an interval not less than one month nor more than three months. The compilation of instrument failure rate data may include data obtained from other Boiling Water Reactors for which the same design instrument operates in an environment similar to that of Dresden Unit 3.

2. Functional test calibrations and instrument checks are not required when these instruments are not required to be operable or are tripped. Functional tests shall be performed before each startup with a required frequency not to exceed once per week. Calibrations shall be performed during each startup or during controlled shutdowns with a required frequency not to exceed once per week. Instrument checks shall be performed at least once per week. Instrument checks shall be performed at least once per day during those periods when the instruments are required to be operable.

3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel. See Note 4.

4. These instrument channels will be calibrated using simulated electrical signals once every three months. In addition, calibration including the sensors will be performed during each refueling outage.

5. A minimum of two channels is required.

6. From and after the date that one of these parameters [... either drywell-torus differential pressure or torus water level indication] is reduced to one indication, continued operation is not permissible beyond thirty days, unless such instrumentation is sooner made operable. In the event that all indications of these parameters [...either drywell-torus differential pressure or torus water level] is disabled and such indication cannot be restored in six (6) hours, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition in twenty four hours.

above. In connection with such testing, the pool temperature must be reduced to below the normal power operation limit specified in (1) above within 24 hours.

(3) The reactor shall be scrammed from any operating condition if the pool temperature reaches 110°F. Power operation shall not be resumed until the pool temperature is reduced below the normal power operation limit specified in (1) above.

(4) During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 150 psig at normal cooldown rates if the pool temperature reaches 120°F.

d. Maximum downcomer submergence is 4.00 ft.

3. Minimum downcomer submergence is 3.67 ft.

2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 Mw(t).

d. A visual inspection of the suppression chamber interior, including water line regions, shall be made at each major refueling outage.

2. The primary containment integrity shall be demonstrated by either Method A or Method B, as follows:

a. Integrated Primary Containment Leak Test (IPCLT)

d. Whenever the reactor is in power operation, the primary containment oxygen sampling system shall be operable. If this specification cannot be met, the system must be restored to an operable condition within 7 days or the reactor must be taken out of power operation.

e. The maximum containment repressurization pressure using the containment makeup inerting system shall be 26 psig.

7. Drywell Suppression Chamber Differential Pressure

A. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.00 psid except as specified in (1), (2), and (3) below:

(1) This differential shall be established within the 24 hour period subsequent to placing the reactor mode switch into the run mode during a startup and may be relaxed 24 hours prior to a reactor shutdown when the provisions of 3.7.A.5 (b) apply.

(2) This differential may be decreased to less than 1.00 psid for a maximum of 4 hours during required operability testing of the drywell pressure suppression chamber vacuum breakers, HPCI testing and reactor pressure relief valve testing.

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

d. The containment oxygen analyzing system shall be functionally tested once per week and shall be calibrated once per 6 months.

7. Drywell Suppression Chamber Differential Pressure

a. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift when the differential pressure is required.

A means to determined post LOCA containment oxygen concentration is necessary to readily enable the reactor operator to take appropriate action to control containment atmosphere. In the interim, prior to installation of the CAD and associated monitoring systems, the containment oxygen analyzing system will be available.

The maximum containment repressurization pressure of 26 psi provides adequate margin to containment design pressure and a delay time prior to purge which results in acceptable purge doses.

Following a LOCA, periodic operation of the drywell and torus sprays will be used to assist the natural convection and diffusion mixing of hydrogen and oxygen when other ECCS requirements are met and O_2 concentration exceeds 4%.

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed (Reference 15) 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 of 1.00 psid 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.

B. Standby Gas Treatment System and C Secondary Containment

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation, when the drywell is sealed and in service; the reactor building provides primary containment when the reactor is shutdown and the drywell is open, as during refueling. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required as well as during refueling.

(15) "Dresden Nuclear Generating Plant Units 2 & 3 Short Term Program Plant Unique Torus Support and Attached Piping Analysis", August 1976 NUTECH Report COM-01-040.

Only one of the two standby gas treatment system circuits is needed to cleanup the reactor building atmosphere upon containment isolation. If one system is found to be inoperable, there is no immediate threat to the containment system performance. Therefore, reactor operation or refueling operation may continue while repairs are being made. If neither circuit is operable, the plant is placed in a condition that does not require a standby gas treatment system.

While only a small amount of particulates are released from the primary containment as a result of the loss of coolant accident, high-efficiency particulate filters before and after the charcoal filters are specified to minimize potential particulate release to the environment and to prevent clogging of the charcoal adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. (The in-place test results should indicate a system leak tightness of less than 1% bypass leakage for the charcoal adsorbers using halogenated hydrocarbon and a HEPA filter efficiency of at least 99% removal of DOP particulates. Laboratory carbon sample test results indicate a radioactive methyl iodide removal efficiency for expected accident conditions. Operation of the standby gas treatment circuits significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performance requirements are met as specified, the calculated doses would be less than the guidelines stated in 10 CFR 100 for the accidents analyzed).*

*Bases in parentheses will not be applicable until about December 31, 1976, when equipment modifications are completed to allow increased testing.

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the stack during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environs. One standby gas treatment fan is designed to automatically start upon containment isolation and to maintain the reactor building pressure to approximately a negative 1/4-inch water guage pressure; all leakage should be in-leakage. Should the fan fail to start, the redundant alternate fan and filter system is designed to start automatically. Each of the two fans has 200% capacity. Ref. Section 5.3.2 SAR. If one standby gas treatment system circuit is inoperable, the other circuit will be tested daily. This substantiates the availability of the operable circuit and results in no added risk; thus, reactor operation or refueling operation can continue. If neither circuit is operable the plant is brought to a condition where the system is not required.

While only a small amount of particulates are released from the pressure suppression chamber system as a result of the loss of coolant accident, high-efficiency particulate filters before and after the charcoal filters are specified to minimize potential particulate release to the environment and to prevent clogging of the charcoal adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The inplace test results should indicate a system leak tightness of less than 1% bypass leakage for the charcoal adsorbers using halogenated hydrocarbon and a HEPA filter efficiency of at least 99% removal of DOP particulates. Laboratory carbon sample test results indicate a radioactive methyl iodide removel efficiency for expected accident conditions. Operation of the standby gas treatment circuits significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performance requirements are met as specified, the calculated doses would be less than the guidelines stated in 10 CFR 100 for the accidents analyzed.

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 STATION UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 46
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 November 30, 1976, as supplemented by letters dated April 18 and April 19, 1977, 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. 46, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 7, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 46

FACILITY OPERATING LICENSE NO. DPR-29

DOCKET NO. 50-254

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by the captioned Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
3.2/4.2-15	3.2/4.2-15
3.2/4.2-18	3.2/4.2-18
3.7/4.7-2	3.7/4.7-2
3.7/4.7-7	3.7/4.7-7
3.7/4.7-7a	3.7/4.7-7a
3.7/4.7/13	3.7/4.7-13
3.7/4.7-13a	----------
3.7/4.7-14	3.7/4.7-14

Add pages 3.2/4.2-15a and 3.7/4.7-6a

TABLE 3.2-4

POSTACCIDENT MONITORING INSTRUMENTATION REQUIREMENTS(2)

Minimum Number of Operable Channels(1)(3)	Parameter	Instrument Readout Location Unit 2	Number Provided	Range
1	Reactor pressure	902-5	1 2	0-1500 psig 0-1200 psig
1	Reactor water level	902-3	2	-100 inches + 200 inches (0 inches is top of fuel)
1	Torus water temperature	902-21	2	0-200°F
1	Torus air temperature	902-21	2	0-600°F
2(4)	Torus water level, indicator	902-3	1	-25 inches - +25 inches
	Torus water level, sight glass		1	18 inch range
1	Torus pressure	902-3	1	-5 inches Hg to 5 psig
1	Drywell pressure	902-3	1	-5inches Hg to 5 psig 0 to 75 psig
2	Drywell temperature	902-21	6	0-600° F
2	Neutron monitoring	902-5	4	0.1-10^6 CPS
2(4)	Torus to drywell differential pressure		2	0-3 psid

Notes

1. Instrument channels required during power operation to monitor postaccident conditions.

2. Provisions are made for local sampling and monitoring of drywell atmosphere.

3. In the event any of the instrumentation becomes inoperable for more than 7 days during reactor operation, initiate an orderly shutdown and be in the cold shutdown condition within 24 hours.

4. From and after the date that one of these parameters is reduced to one indication, continued operation is not permissible beyond thirty days, unless such instrumentation is sooner made operable. In the event that all indication of these parameters is disabled and such indication cannot be restored in six (6) hours, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition in twenty four (24) hours.

TABLE 4.2-2

POSTACCIDENT MONITORING INSTRUMENTATION REQUIREMENTS

Minimum Number of Operable Channels*	Parameter	Instrument Readout Location Unit 2	Calibration	Instrument check
1	Reactor pressure	902-5	Once every 3 months	Once per day
1	Reactor water level	902-3	Once every 3 months	Once per day
1	Torus water temperature	902-21	Once every 3 months	Once per day
1	Torus air temperature	902-21	Once every 3 months	Once per day
2	Torus water level (indicator)	902-3	Once every 3 months	Once per day
	Torus water level (sight glass)		N/A	None
1	Torus pressure	902-3	Once every 3 months	Once per day
1	Drywell pressure	902-3	Once every 3 months	Once per day
2	Drywell temperature	902-21	Once every 3 months	Once per day
2	Neutron monitoring	902-5	Once every 3 months	Once per day
2	Torus to drywell differential pressure		Once every 6 months	None

*Instrument channels required during power operation to monitor postaccident conditions.

power operation limit specified in Item 1 within 24 hours.

3) The reactor shall be scrammed from any operating condition if the pool temperature reaches 110° F. Power operation shall not be resumed until the pool temperature is reduced below the normal power operation limit specified in Item 1.

4) During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 150 psig at normal cooldown rates if the pool temperature reaches 120° F.

d. Maximum downcomer Submergence - 3.54 ft.

e. Minimum downcomer Submergence - 3.21 ft.

2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212° F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 MWt.

2. The primary containment integrity shall be demonstrated by conducting integrated primary containment leak tests (IPCLT).

   a. IPCLT shall be performed at an initial pressure of approximately 48 psig, P_t (48).

   b. If local leak rate measurements are made prior to IPCLT and repairs are found to be necessary and retests conducted, the leak rate difference prior to and after repair when corrected to P_t (48) shall be added to the final integrated leak rate result.

   c. Closure of the containment isolation valves for the purpose of the test shall be accomplished by the means provided for normal operation of the valves.

   d. The test duration shall not be less than 24 hours for integrated leak rate measurements, but shall be extended to a sufficient period of time to verify, by measuring the quantity of air required to return to the starting point (or other methods of equivalent sensitivity), the validity and accuracy of the leak rate results.

except as specified in Specification 3.7.A.5.b.

b. Within the 24-hour period subsequent to placing the reactor in the Run mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 5% by weight, and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.

6. Containment Systems

Drywell-Suppression Chamber Differential Pressure

a. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.20 psid except as specified in (1), (2), and (3) below:

(1) This differential shall be established within the 24 hour period subsequent to placing the reactor mode switch into the RUN mode during a startup and may be relaxed 24 hours prior to reactor shutdown when the provisions of 3.7.A.5(b) apply.

(2) This 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.

6. CONTAINMENT SYSTEMS

Drywell-Suppression Chamber Differential Pressure

a. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift.

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

7.

B. Standby Gas Treatment System

1. Two separate and independent standby gas treatment system 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 system circuits 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 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 ($\pm$ 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 seven days, initiate from the control room 4000 cfm ($\pm$ 10%) flow through the operable circuit of the standby gas treatment system for at least 10 hours with the circuit heaters operating.

hydrogen, if it is present in sufficient quantities to result in excessively rapid recombination, could result in a loss of containment integrity.

The 5% oxygen concentration minimizes the possiblity of hydrogen combustion following a loss-of-coolant accident. Significant quantities of hydrogen could be generated if the core cooling systems did not sufficiently cool the core.

The occurrence of primary system leakage following a major refueling outage or other scheduled shutdown is much more probable than the occurrence of the loss-of-coolant accident upon which the specified oxygen concentration limit is based. Permitting access to the drywell for leak inspections during a startup is judged prudent in terms of the added plant safety offered without significantly reducing the margin of safety. Thus, to preclude the possibility of starting the reactor and operating for extended periods of time with significant leaks in the primary system, leak inspections are scheduled during startup periods, when the primary system is at or near rated temperature and pressure.

The 24-hour period to provide inerting is judged to be sufficient to perform the leak inspection and establish the required oxygen concentration. The primary containment is normally slightly pressurized during periods of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase oxygen concentration. Once the containment is filled with nitrogen to the required concentration, no monitoring of oxygen concentration is necessary. However, at least once a week, the oxygen concentration will be determined as added assurance.

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed (Reference 5) 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 of 1.20 psid and a suppression chamber water level corresponding to a downcomer submergence range of 3.21 to 3.54 feet will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

B. Standby Gas Treatment System

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the chimney during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environs. One standby gas treatment system circuit is designed to automatically start upon containment isolation and to maintain the reactor building pressure at the design negative pressure so that all leakage should be in-leakage. Should one circuit fail to start, the redundant alternate standby gas treatment circuit is designed to start automatically. Each of the two circuits has 100% capacity. Only one of the two standby gas treatment system circuits is needed to cleanup the reactor building atmosphere upon containment isolation. If one system is found to be inoperable, there is not immediate threat to the containment system performance. Therefore, reactor operation or refueling operation may continue while repairs are being made. If neither circuit is operable, the plant is placed in a condition that does not require a standby gas treatment system.

While only a small amount of particulates are released from the primary containment as a result of the loss-of-coolant accident, high-efficiency particulate filters before and after the charcoal filters are specified to minimize potential particulate release to the environment and to prevent clogging of the charcoal adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. (The in-place test results should indicate a system leak tightness of less than 1% bypass leakage for the charcoal adsorbers using halogenated hydrocarbon and a HEPA filter efficiency of at least 99% removal of DOP particulates.

Laboratory carbon sample test results indicate a radioactive methyl iodine removal efficiency for expected accident conditions. Operation of the standby gas treatment circuits significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performance requirements are met as specified, the calculated doses would be less than the guidelines stated in 10 CFR 100 for the accidents analyzed.

C. Secondary Containment

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation, when the drywell is sealed and in service; the reactor building provides primary containment when the reactor is shut down and the drywell is open, as during refueling. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required as well as during refueling, except, however, for initial fuel loading of Unit 1 prior to initial power testing (reference SAR Section 1).

D. Primary Containment Isolation Valves

Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident.

References

1. 'Bodega Bay Preliminary Hazards Summary Report.' Appendix 1, Docket 50-205, December 28, 1962.
2. C. H. Robbins, 'Tests of a Full Scale 1/48 Segment of the Humboldt Bay Pressure Suppression Containment,' GEAP-3596, November 17, 1960.
3. Quad-Cities Special Report Number 4.
4. 'Nuclear Safety Program Annual Progress Report for Period Ending December 31, 1966, ORNL-4071.'
5. "Quad-Cities Station Units 1 and 2 Short Term Program Plant Unique Torus Support and Attached Piping Analyses" August 1976 NUTECH Report Number COM-01-039.

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 STATION UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 46
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 November 30, 1976, as supplemented by letters dated April 18 and April 19, 1977, 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. 46, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 7, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 46

FACILITY OPERATING LICENSE NO. DPR-30

DOCKET NO. 50-265

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by the captioned Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
3.2/4.2-15	3.2/4.2-15
3.2/4.2/18	3.2/4.2/18
3.7/4.7-2	3.7/4.7-2
3.7/4.7-7	3.7/4.7-7
3.7/4.7-13	3.7/4.7-13
3.7/4.7-13a	----------
3.7/4.7-14	3.7/4.7-14

Add pages 3.2/4.2-15a and 3.7/4.7-6a

TABLE 3.2-4

POSTACCIDENT MONITORING INSTRUMENTATION REQUIREMENTS (2)

Minimum Number of Operable Channels (1)(3)	Parameter	Instrument Readout Location Unit 2	Number Provided	Range
1	Reactor pressure	902-5	1 2	0-1500 psig 0-1200 psig
1	Reactor water level	902-3	2	-100 inches + 200 inches (0 inches is top of fuel)
1	Torus water temperature	902-21	2	0-200°F
1	Torus air temperature	902-21	2	0-600°F
2 (4)	Torus water level, indicator	902-3	1	-25 inches - +25 inches
	Torus water level, sight glass		1	18 inch range
1	Torus pressure	902-3	1	-5 inches Hg to 5 psig
1	Drywell pressure	902-3	1	-5inches Hg to 5 psig 0 to 75 psig
2	Drywell temperature	902-21	6	0-600° F
2	Neutron monitoring	902-5	4	$0.1\text{-}10^6$ CPS
2 (4)	Torus to drywell differential pressure		2	0-3 psid

Notes

1. Instrument channels required during power operation to monitor postaccident conditions.

2. Provisions are made for local sampling and monitoring of drywell atmosphere.

3. In the event any of the instrumentation becomes inoperable for more than 7 days during reactor operation, initiate an orderly shutdown and be in the cold shutdown condition within 24 hours.

4. From and after the date that one of these parameters is reduced to one indication, continued operation is not permissible beyond thirty days, unless such instrumentation is sooner made operable. In the event that all indication of these parameters is disabled and such indication cannot be restored in six (6) hours, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition in twenty four (24) hours.

TABLE 4.2-2

POSTACCIDENT MONITORING INSTRUMENTATION REQUIREMENTS

Minimum Number of Operable Channels*	Parameter	Instrument Readout Location Unit 2	Calibration	Instrument check
1	Reactor pressure	902-5	Once every 3 months	Once per day
1	Reactor water level	902-3	Once every 3 months	Once per day
1	Torus water temperature	902-21	Once every 3 months	Once per day
1	Torus air temperature	902-21	Once every 3 months	Once per day
2	Torus water level (indicator)	902-3	Once every 3 months	Once per day
	Torus water level (sight glass)		N/A	None
1	Torus pressure	902-3	Once every 3 months	Once per day
1	Drywell pressure	902-3	Once every 3 months	Once per day
2	Drywell temperature	902-21	Once every 3 months	Once per day
2	Neutron monitoring	902-5	Once every 3 months	Once per day
2	Torus to drywell differential pressure		Once every 6 months	None

*Instrument channels required during power operation to monitor postaccident conditions.

power operation limit specified in Item 1 within 24 hours.

3) The reactor shall be scrammed from any operating condition if the pool temperature reaches 110° F. Power operation shall not be resumed until the pool temperature is reduced below the normal power operation limit specified in Item 1.

4) During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 150 psig at normal cooldown rates if the pool temperature reaches 120° F.

d. Maximum downcomer Submergence - 3.54 ft.

e. Minimum downcomer Submergence - 3.21 ft.

2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212° F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 MWt.

2. The primary containment integrity shall be demonstrated by conducting integrated primary containment leak tests (IPCLT).

a. IPCLT shall be performed at an initial pressure of approximately 48 psig, P_t (48).

b. If local leak rate measurements are made prior to IPCLT and repairs are found to be necessary and retests conducted, the leak rate difference prior to and after repair when corrected to P_t (48) shall be added to the final integrated leak rate result.

c. Closure of the containment isolation valves for the purpose of the test shall be accomplished by the means provided for normal operation of the valves.

d. The test duration shall not be less than 24 hours for integrated leak rate measurements, but shall be extended to a sufficient period of time to verify, by measuring the quantity of air required to return to the starting point (or other methods of equivalent sensitivity), the validity and accuracy of the leak rate results.

except as specified in Specification 3.7.A.5.b.

b. Within the 24-hour period subsequent to placing the reactor in the Run mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 5% by weight, and maintained in this condition. Deinerting may commence 24 hours prior to a shutdown.

6. Containment Systems

Drywell-Suppression Chamber Differential Pressure

a. Differential pressure between the drywell and suppression chamber shall be maintained at equal to or greater than 1.20 psid except as specified in (1), (2), and (3) below:

(1) This differential shall be established within the 24 hour period subsequent to placing the reactor mode switch into the RUN mode during a startup and may be relaxed 24 hours prior to reactor shutdown when the provisions of 3.7.A.5(b) apply.

(2) This 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.

6. CONTAINMENT SYSTEMS

Drywell-Suppression Chamber Differential Pressure

a. The pressure differential between the drywell and suppression chamber shall be recorded at least once each shift.

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

7.

B. Standby Gas Treatment System

1. Two separate and independent standby gas treatment system 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 system circuits 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 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 ($\pm$ 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 seven days, initiate from the control room 4000 cfm ($\pm$ 10%) flow through the operable circuit of the standby gas treatment system for at least 10 hours with the circuit heaters operating.

hydrogen, if it is present in sufficient quantities to result in excessively rapid recombination, could result in a loss of containment integrity.

The 5% oxygen concentration minimizes the possiblity of hydrogen combustion following a loss-of-coolant accident. Significant quantities of hydrogen could be generated if the core cooling systems did not sufficiently cool the core.

The occurrence of primary system leakage following a major refueling outage or other scheduled shutdown is much more probable than the occurrence of the loss-of-coolant accident upon which the specified oxygen concentration limit is based. Permitting access to the drywell for leak inspections during a startup is judged prudent in terms of the added plant safety offered without significantly reducing the margin of safety. Thus, to preclude the possibility of starting the reactor and operating for extended periods of time with significant leaks in the primary system, leak inspections are scheduled during startup periods, when the primary system is at or near rated temperature and pressure.

The 24-hour period to provide inerting is judged to be sufficient to perform the leak inspection and establish the required oxygen concentration. The primary containment is normally slightly pressurized during periods of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase oxygen concentration. Once the containment is filled with nitrogen to the required concentration, no monitoring of oxygen concentration is necessary. However, at least once a week, the oxygen concentration will be determined as added assurance.

In conjunction with the Mark I Containment Short Term Program, a plant unique analysis was performed (Reference 5) 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 of 1.20 psid and a suppression chamber water level corresponding to a downcomer submergence range of 3.21 to 3.54 feet will assure the integrity of the suppression chamber when subjected to post-LOCA suppression pool hydrodynamic forces.

B. Standby Gas Treatment System

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the chimney during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environs. One standby gas treatment system circuit is designed to automatically start upon containment isolation and to maintain the reactor building pressure at the design negative pressure so that all leakage should be in-leakage. Should one circuit fail to start, the redundant alternate standby gas treatment circuit is designed to start automatically. Each of the two circuits has 100% capacity. Only one of the two standby gas treatment system circuits is needed to cleanup the reactor building atmosphere upon containment isolation. If one system is found to be inoperable, there is not immediate threat to the containment system performance. Therefore, reactor operation or refueling operation may continue while repairs are being made. If neither circuit is operable, the plant is placed in a condition that does not require a standby gas treatment system.

While only a small amount of particulates are released from the primary containment as a result of the loss-of-coolant accident, high-efficiency particulate filters before and after the charcoal filters are specified to minimize potential particulate release to the environment and to prevent clogging of the charcoal adsorbers. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. (The in-place test results should indicate a system leak tightness of less than 1% bypass leakage for the charcoal adsorbers using halogenated hydrocarbon and a HEPA filter efficiency of at least 99% removal of DOP particulates.

Laboratory carbon sample test results indicate a radioactive methyl iodine removal efficiency for expected accident conditions. Operation of the standby gas treatment circuits significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers. If the performance requirements are met as specified, the calculated doses would be less than the guidelines stated in 10 CFR 100 for the accidents analyzed.

C. Secondary Containment

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation, when the drywell is sealed and in service; the reactor building provides primary containment when the reactor is shut down and the drywell is open, as during refueling. Because the secondary containment is an integral part of the complete containment system, secondary containment is required at all times that primary containment is required as well as during refueling, except, however, for initial fuel loading of Unit 1 prior to initial power testing (reference SAR Section 1).

D. Primary Containment Isolation Valves

Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident.

References

1. 'Bodega Bay Preliminary Hazards Summary Report.' Appendix 1, Docket 50-205, December 28, 1962.
2. C. H. Robbins, 'Tests of a Full Scale 1/48 Segment of the Humboldt Bay Pressure Suppression Containment,' GEAP-3596, November 17, 1960.
3. Quad-Cities Special Report Number 4.
4. 'Nuclear Safety Program Annual Progress Report for Period Ending December 31, 1966, ORNL-4071.'
5. "Quad-Cities Station Units 1 and 2 Short Term Program Plant Unique Torus Support and Attached Piping Analyses" August 1976 NUTECH Report Number COM-01-039.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

AMENDMENT NOS. 37, 35, 46 AND 46

LICENSE NOS. DPR-19, DPR-25, DPR-29, AND DPR-30

DRESDEN AND QUAD CITIES NUCLEAR POWER STATIONS

COMMONWEALTH EDISON COMPANY

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

1. INTRODUCTION

In conjunction with the Short Term Program (STP) evaluation of Boiling Water Reactor facilities with the Mark I containment system, the Commonwealth Edison Company (the licensee) submitted a Plant Unique Analysis (PUA) for Dresden Station Unit Nos. 2 and 3 and Quad Cities Unit Nos. 1 and 2. These analyses were performed to confirm the structural and functional capability of the containment suppression chamber and attached piping, to withstand newly-identified suppression pool hydrodynamic loading conditions which had not been explicitly considered in the original design analysis for the plant. As part of the STP evaluation, specific loading conditions were developed for each Mark I facility, to account for the change in the magnitude of the loads due to plant-specific variations from the reference plant design for which the basic loading conditions were developed.

The results of the NRC staff's review of the hydrodynamic load definition techniques and the Mark I containment plant unique analyses are described in the "Mark I Containment Short Term Program Safety Evaluation Report," NUREG-0408, December 1977. As discussed in this report, the NRC staff has concluded that each Mark I containment system would maintain its integrity and functional capability in the unlikely event of a design basis loss-of-coolant accident (LOCA) and, therefore, that licensed Mark I BWR facilities can continue to operate safely, without undue risk to the health and safety of the public, during an interim period of approximately two years, while a methodical, comprehensive Long Term Program is conducted.

As discussed in Section III.C of NUREG-0408, of all of the plant parameters that were considered in the development of the hydrodynamic loads for the STP, only two parameters are expected to vary during normal plant operation; these are (1) the drywell-wetwell differential pressure; and (2) the suppression chamber (torus) water level. Subsequent to the submittal of the PUA, the licensee was requested to submit proposed Technical Specifications which assure that the allowable range of these two parameters during facility operation would be in accordance with the values utilized in the PUA.

The licensee has been operating these facilities with differential pressure control to enhance the safety margins of the containment structure since early 1976. This evaluation provides a more detailed basis for establishing the allowable range of drywell-wetwell differential pressure and torus water levels, in order to quantify containment safety margins. This amendment incorporates these parameters into the Technical Specifications with the associated limiting condition and surveillance requirements.

By letters dated November 30, 1976, April 18 and April 19, 1977, the licensee proposed changes to the facility Technical Specifications to incorporate limiting conditions for operation and surveillance requirements for differential pressure control and torus water level. Our evaluation of these proposed changes follows.

II. EVALUATION

The licensee has proposed certain Technical Specification requirements for the purpose of assuring that the normal plant operating conditions are within the envelope of conditions considered in their PUA. These Technical Specification changes establish (1) limiting condition for operation (LCOs) for drywell to torus differential pressure and torus water level, and (2) associated surveillance requirements. All other initial conditions utilized in the PUA are either presently included in the Technical Specifications or are configurational conditions which have been confirmed by the licensee and will not change during normal operation.

Differential pressure between the drywell and the suppression chamber will result in leakage of the drywell atmosphere to the lower pressure regions of the reactor building and to the torus airspace. This leakage from the drywell will cause a slow decay in the differential pressure. Therefore, surveillance requirements for the differential pressure have been included in the Technical Specifications. Surveillance frequency of once per operating shift for the differential pressure was selected on the basis of previous operating experience.

The torus water level is not expected to vary significantly during normal operation, unless certain systems connected to the suppression pool are activated. The torus water level would normally be monitored whenever such systems are in use. Therefore, we find that inclusion of periodic torus water level surveillance requirements in the Technical Specifications is not required.

We have reviewed the differential pressure and torus water level monitoring instrumentation systems proposed by the licensee with regard to the number of available channels and the instrumentation accuracy. This type of instrumentation is typically calibrated at six-month intervals. To assure proper operation during such intervals, two monitoring channels for both differential pressure and torus water level have been provided such that a comparision of the readings will indicate when one of the channels is inoperative or drifting.

inoperative or drifting. The calibration frequency for differential pressure instrumentation has been reduced to once per month, because only one channel of drywell-torus differential pressure has been provided. Based on extensive operating experience with pressure transducers, we conclude that a calibration frequency of once per month will provide adequate assurance of the continued accuracy of the drywell-torus differential pressure measurement.

The errors in the instrumentation are sufficiently small relative to the magnitude of the measurement (i.e., a maximum differential pressure measurement error of 0.1 psid in a measurement of 1.0 to 2.0 psid and a maximum torus water level measurement error of 10% of the difference between the maximum and minimum torus water level) that they may be neglected, based on the expected load variation with differential pressure and torus water level.

There are certain periods during normal plant operations when the differential pressure control cannot be maintained. Therefore, provisions have been included in the Technical Specifications to relax the differential pressure/control requirements during specified periods. The justification for relaxing the differential pressure control during these specific periods and the basis for selecting the duration of the periods are discussed in detail below.

A. Startup and Shutdown

During plant startup and shutdown, the drywell atmosphere undergoes significant barometric changes due to the variation in heat loads from the primary and auxiliary systems. In addition, it is during these periods that the drywell is being either inerted with nitrogen gas or deinerted. In order to keep the periods during which the differential pressure control is not fully effective as short as is reasonable, we have limited the relaxation of the differential pressure control requirements for the startup and shutdown periods to 24 hours following startup and 24 hours prior to a shutdown. This time period was selected on a basis similar to that for the inerting requirements, already existing in the Technical Specifications. The postulated design basis accident for the containment assumes that the primary system is at operating pressure and temperature. During the startup and shutdown transients, the primary system is at operating pressure and temperature for only a part of the transient, during which the differential pressure is being established. These time periods have been shown by previous operating experience to be adequate with respect to the startup and shutdown transients, and at the same time sufficiently small in comparison to the duration of the average power run. Since the principal accident event for which differential pressure control is important to assure containment integrity, (i.e., with a factor of safety of two) is a large break LOCA, we have considered whether there is a significantly greater probability of a large break LOCA during the startup and shutdown transients. We have concluded that there is not. Further, the operation of the plant systems is monitored more closely than normal during these periods and a finite magnitude of differential pressure will be available during the majority of these periods to mitigate the potential consequences of an accident.

B. Testing and Maintenance

During normal operation, there are a number of tests which are required to be conducted to demonstrate the continued functional performance of engineered safety features. The testing of certain systems will require, or result in, a reduction in the drywell-torus differential pressure. The operability testing of the drywell-torus vacuum breakers requires the removal of the differential pressure and permit the vacuum breakers to open. For the testing of high-energy systems (e.g., high pressure coolant injection pumps) during normal operation, the discharge flow is routed to the suppression pool. This energy deposition will raise the temperature of the suppression pool, resulting in an increase in torus pressure and a reduction in the differential pressure.

Functional performance testing of engineered safety features is necessary to assure proper maintenance of these systems throughout the life of the plant. Some of these tests (i.e., pump operability and drywell-wetwell vacuum breakers) may require or result in a reduction in the differential pressure. We estimate that not more than four tests will be required each month which will result in a reduction in differential pressure. In order to keep the periods during which the differential pressure control is not fully effective as short as is reasonable, we have permitted a relaxation of differential pressure control in order to conduct these tests, limited to a period of up to four hours. Again, we have carefully considered whether the probability of a large LOCA is significantly greater during these testing periods than that during normal operation. We conclude that it is not. Moreover, only the test of the drywell-wetwell vacuum breakers requires complete removal of the differential pressure.

Provisions have also been included in the Technical Specifications for performing maintenance activities on the differential pressure control system and for resolving operational difficulties which may result in an inadvertent reduction in the differential pressure for a short period of time. In certain circumstances, corrective action can be taken without having to attain a cold shutdown condition. To avoid repeated and unnecessary partial cooldown cycles, a restoration period has been incorporated into the action requirements of the LCO for differential pressure control; i.e., in the event that the differential pressure cannot be restored in six hours, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours. The six hour restoration period was selected on the basis that it represents an adequate minimum period of time during which any short-term malfunctions could be corrected, coupled with the minimum period of time required to conduct a controlled shutdown.

The allowable time to conduct a controlled shutdown has been minimized, because the containment transient response is more a function of the primary system pressure than the reactor power level. On this basis, we find the proposed restoration period and action requirement acceptable.

We conclude that the limits imposed on the periods of time during which operation is permitted without the differential pressure control fully effective provides adequate assurance of overall containment integrity, and the periods of time differential pressure control is completely removed are acceptably small.

ENVIRONMENTAL CONSIDERATION

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant enviornmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR Section 51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

CONCLUSIONS

The proposed Technical Specifications will provide the necessary assurance that the plant's operating conditions remain within the envelope of the conditions assumed in the Plant Unique Analysis (PUA) performed in conjunction with the Mark I Containment Short Term Program. The PUA supplements the facility's Final Safety Analysis Report (FSAR) in that it demonstrates the plant's capability to withstand the suppression pool hydrodynamic loads which were not explicitly considered in the FSAR. We therefore conclude that the proposed changes to the Technical Specifications are acceptable.

We further conclude, based on the considerations discussed above, that (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated:

UNITED STATES NUCLEAR REGULATORY COMMISSION

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

COMMONWEALTH EDISON COMPANY
AND
IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued an amendment each to Facility Operating License Nos. DPR-19, DPR-25, DPR-29 and DPR-30, issued to Commonwealth Edison Company (and, in the matter of License Nos. DPR-29 and DPR-30, the Iowa-Illinois Gas and Electric Company), which revised Technical Specifications for operation of each of the Dresden and Quad Cities Nuclear Power Stations (collectively referred to as the facilities). The Dresden Station consists of Unit Nos. 1, 2, and 3 and is located in Grundy County, Illinois. However, the actions noticed herein relate to Dresden Station Units 2 and 3. The Quad Cities Station consists of Unit Nos. 1 and 2 and is located in Rock Island County, Illinois. These amendments are effective as of their dates of issuance.

The amendments revised the Technical Specifications to incorporate requirements for establishing and maintaining the drywell to suppression chamber differential pressure and suppression chamber water level, to maintain the margins of safety established in the Commission staff's "Mark I Containment Short Term Program Safety Evaluation," NUREG-0408. Operation in accordance with the conditions specified in NUREG-0408 has been previously authorized in 43 FR 13106 March 29, 1978, and 43 FR 77415, April 24, 1978.

The application for the amendments comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR Section 51.5(d)(4), an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of the amendments.

For further details with respect to this action, see (1) the application for amendment dated November 30, 1976 and supplements thereto dated April 18 and April 19, 1977, (2) Amendment Nos. 37 and 35 to License Nos. DPR-19, and DPR-25, (3) Amendment Nos. 46 and 46 to License Nos. DPR-29 and DPR-30, and (4) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and for those items relating to Dresden Unit Nos. 2 and 3 at the Morris Public Library, 604 Liberty Street, Morris, Illinois 60450 and for those items relating to Quad Cities Unit Nos. 1 and 2 at the Moline Public Library, 504 - 17th Street, Moline, Illinois 60625. A single copy of items (2), (3), and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland this 7th day of June 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors