April 1, 1982

Docket Nos. 50-237
50-249

Mr. L. DelGeorge
Director of Nuclear Licensing
Commonwealth Edison Company
P.O. Box 767
Chicago, Illinois 60690

Dear Mr. DelGeorge:

The Commission has issued the enclosed Amendment No. 68 to Provisional Operating License No. DPR-19 for Dresden Nuclear Power Station Unit 2 and Amendment No. 60 to Facility Operating License No. DPR-25 for Dresden Nuclear Power Station Unit 3. These amendments consist of changes to the Technical Specifications in response to your application dated November 30, 1981.

The changes provide for primary containment integrated leak rate test requirements and schedules consistent with Appendix J to 10 CFR Part 50. The changes also provide for direct references to and use of Appendix J methodology and terminology.

Copies of our Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

Joseph D. Hegner, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 68 to DPR-19
2. Amendment No. 60 to DPR-25
3. Safety Evaluation
4. Notice

cc: w/enclosures
See next page

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Mr. L. DelGeorge

cc
Isham, Lincoln & Beale
Counselors at Law
One First National Plaza, 42nd Floor
Chicago, Illinois 60603

Plant Superintendent
Dresden Nuclear Power Station
Rural Route #1
Morris, Illinois 60450

U. S. Nuclear Regulatory Commission
Resident Inspectors Office
Dresden Station
RR #1
Morris, Illinois 60450

Mary Jo Murray
Assistant Attorney General
Environmental Control Division
188 W. Randolph Street
Suite 2315
Chicago, Illinois 60601

Morris Public Library
604 Liberty Street
Morris, Illinois 60451

Chairman
Board of Supervisors of
Grundy County
Grundy County Courthouse
Morris, Illinois 60450

John F. Wolf, Esquire
3409 Shepherd Street
Chevy Chase, Maryland 20015

Dr. Linda W. Little
500 Hermitage Drive
Raleigh, North Carolina 27612

Illinois Department of Nuclear Safety
1035 Outer Park Drive, 5th Floor
Springfield, Illinois 62704

U. S. Environmental Protection Agency
Federal Activities Branch
Region V Office
ATTN: Regional Radiation Representative
230 South Dearborn Street
Chicago, Illinois 60604

Dr. Forrest J. Remick
305 East Hamilton Avenue
State College, Pennsylvania 16801

The Honorable Tom Corcoran
United States House of Representatives
Washington, D. C. 20515

James G. Keppler
Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL 60137

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION UNIT 2

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 68
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 November 30, 1981 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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 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. 68, 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

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 1, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 68

PROVISIONAL OPERATING LICENSE NO. DPR-19

DOCKET NO. 50-237

Revise the Appendix "A" Technical Specifications as follows:

Remove	Replace
108a	108a
109	109
110	110
111	111-115*
112	
113	
114	
115	
130	130

*Pages 111-115 are now intentionally blank.

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.

e. Minimum downcomer submergence is 3.67 ft.

f. If specifications 3.7.A.1.a or 3.7.A.1.b are not met and suppression pool water volume 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 within 24 hours.

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

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).

a. Primary containment leakage rates are defined from:

(1) The calculated peak containment internal pressure, P_a, is equal to 48 psig.

(2) The containment vessel reduced test pressure, P_t, is equal to 25 psig.

(3) The maximum allowable leakage rate at a pressure of P_a, L_a, is equal to 1.6 percent by weight of the containment air per 24 hours at 48 psig.

(4) The maximum allowable test leakage rate at a pressure of P_t, L_t, is less than or equal to L_a (L_{tm}/L_{am}). If L_{tm}/L_{am} is greater than 0.7, L_t is (specified as equal to) L_a $(P_t/P_a)^{\frac{1}{2}}$.

(5) The total measured leakage rates at pressures of P_a and P_t are L_{am} and L_{tm}, respectively.

b. When primary containment integrity is required, primary containment leakage rates shall be limited to:

(1) An overall integrated leakage rate for Type A tests of:

(a) L_{am} less than or equal to 75 percent of L_a.

(b) L_{tm} less than or equal to 75 percent of L_t.

2. The primary containment integrity shall be demonstrated by conducting Primary Containment Leak Tests and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and references therein.

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at approximately equal intervals during each 10 year plant in-service inspection interval at either P_a or P_t with the last being done during the 10-year in-service inspection shutdown.

b. If any periodic Type A test fails to meet either 75 percent of L_a or 75 percent of L_t, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission.

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 between the supplemental data and the imposed leakage is within 25 percent of L_a or 25 percent of L_t.

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

(2) (a) A combined leakage rate of less than or equal to 60 percent of L_a for all testable penetrations and isolation valves subject to Type B and C tests except for main steam isolation valves.

(b) A leakage rate of less than or equal to 3.75 percent of L_a for any one air lock when pressurized to 10 psig.

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

(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 or P_t.

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

(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 which shall be tested at 10 psig each operating cycle.

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.

(2) This monitoring system may be taken out of service for the purpose of maintenance or testing but shall be returned to service as soon as practical.

g. The interior surfaces of the drywell shall be visually inspected each operating cycle for evidence of deterioration.

Pages 111-115 are intentionally blank.

The maximum allowable test leak rate is 1.6%/day at a pressure of 48 psig. This value for the test condition was derived from the maximum allowable accident leak rate of about 2.0%/day when corrected for the effects of containment environment under accident and test conditions. In the accident case, the containment atmosphere initially would be composed of steam and hot air whereas under test conditions the test medium would be air or nitrogen at ambient conditions. Considering the differences in mixture composition and temperatures, the appropriate correction factor applied was 0.8 and determined from the guide on containment testing(13).

Although the dose calculations suggest that the accident leak rate could be allowed to increase to about 3.2%/day before the guideline thyroid dose value given in 10 CFR 100 would be exceeded, establishing the test limit of 1.6%/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 or the allowable test 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. The leak rate frequency is based on the AEC guide for developing leak rate testing and surveillance of reactor containment vessels(14). 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 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. The personnel air lock is tested at 10 psig, because the inboard door is not designed to shut in the opposite direction.

The results of the loss-of-coolant accident analyses presented in Amendment No. 18 of the

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(13) TID 20583, Leakage Characteristics of Steel Containment Vessel and the Analysis of Leakage Rate Determinations.

(14) Technical Safety Guide, "Reactor Containment Leakage Testing and Surveillance Requirements" USAEC, Division of Safety Standards, Revised Draft, December 15, 1966.

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-249

DRESDEN NUCLEAR POWER STATION UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 60
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, 1981 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's 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. 60, 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

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 1, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 60

FACILITY OPERATING LICENSE NO. DPR-25

DOCKET NO. 50-249

Revise the Appendix "A" Technical Specifications as follows:

Remove	Replace
108a	108a
109	109
110	110
111	111-115*
112	
113	
114	
115	
130	130

*Pages 111-115 are now intentionally blank.

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.

e. Minimum downcomer submergence is 3.67 ft.

f. If specifications 3.7.A.1.a or 3.7.A.1.b are not met and suppression pool water volume 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 within 24 hours.

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

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).

   a. Primary containment leakage rates are defined from:

      (1) The calculated peak containment internal pressure, P_a, is equal to 48 psig.

      (2) The containment vessel reduced test pressure, P_t, is equal to 25 psig.

      (3) The maximum allowable leakage rate at a pressure of P_a, L_a, is equal to 1.6 percent by weight of the containment air per 24 hours at 48 psig.

      (4) The maximum allowable test leakage rate at a pressure of P_t, L_t, is less than or equal to L_a (L_{tm}/L_{am}). If L_{tm}/L_{am} is greater than 0.7, L_t is (specified as equal to) L_a $(P_t/P_a)^{\frac{1}{2}}$.

      (5) The total measured leakage rates at pressures of P_a and P_t are L_{am} and L_{tm}, respectively.

   b. When primary containment integrity is required, primary containment leakage rates shall be limited to:

      (1) An overall integrated leakage rate for Type A tests of:

         (a) L_{am} less than or equal to 75 percent of L_a.

         (b) L_{tm} less than or equal to 75 percent of L_t.

2. The primary containment integrity shall be demonstrated by conducting Primary Containment Leak Tests and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and references therein.

   a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at approximately equal intervals during each 10 year plant in-service inspection interval at either P_a or P_t with the last being done during the 10-year in-service inspection shutdown.

   b. If any periodic Type A test fails to meet either 75 percent of L_a or 75 percent of L_t, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission.

   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 between the supplemental data and the imposed leakage is within 25 percent of L_a or 25 percent of L_t.

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

3.7 LIMITING CONDITION FOR OPERATION

(2) (a) A combined leakage rate of less than or equal to 60 percent of L_a for all testable penetrations and isolation valves subject to Type B and C tests except for main steam isolation valves.

(b) A leakage rate of less than or equal to 3.75 percent of L_a for any one air lock when pressurized to 10 psig.

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

4.7 SURVEILLANCE REQUIREMENTS

(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 or P_t.

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

(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 which shall be tested at 10 psig each operating cycle.

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.

(2) This monitoring system may be taken out of service for the purpose of maintenance or testing but shall be returned to service as soon as practical.

g. The interior surfaces of the drywell shall be visually inspected each operating cycle for evidence of deterioration.

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The maximum allowable test leak rate is 1.6%/day at a pressure of 48 psig. This value for the test condition was derived from the maximum allowable accident leak rate of about 2.0%/day when corrected for the effects of containment environment under accident and test conditions. In the accident case, the containment atmosphere initially would be composed of steam and hot air whereas under test conditions the test medium would be air or nitrogen at ambient conditions. Considering the differences in mixture composition and temperatures, the appropriate correction factor applied was 0.8 and determined from the guide on containment testing(13).

Although the dose calculations suggest that the accident leak rate could be allowed to increase to about 3.2%/day before the guideline thyroid dose value given in 10 CFR 100 would be exceeded, establishing the test limit of 1.6%/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 or the allowable test 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. The leak rate frequency is based on the AEC guide for developing leak rate testing and surveillance of reactor containment vessels(14). 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 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. The personnel air lock is tested at 10 psig, because the inboard door is not designed to shut in the opposite direction.

The results of the loss-of-coolant accident analyses presented in Amendment No. 18 of the

(13) TID 20583, Leakage Characteristics of Steel Containment Vessel and the Analysis of Leakage Rate Determinations.

(14) Technical Safety Guide, "Reactor Containment Leakage Testing and Surveillance Requirements" USAEC, Division of Safety Standards, Revised Draft, December 15, 1966.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 68 TO PROVISIONAL OPERATING LICENSE NO. DPR-19

AND AMENDMENT NO. 60 TO FACILITY OPERATING LICENSE NO. DPR-25

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION, UNTIS 2 AND 3

DOCKET NOS. 50-237 AND 50-249

Author: J. D. Hegner

I. INTRODUCTION

By letter dated November 30, 1981 Commonwealth Edison Company (licensee) proposed changes to the Technical Specifications for Dresden Nuclear Power Station, Units 2 and 3 to: revise the primary containment integrated leak rate test requirements and schedules to conform with the requirements of Appendix J to 10 CFR Part 50; modify the associated Limiting Condition for Operation to include the definitions of the nomenclature used and identify specific leakage limitations as required by Appendix J; and modify the surveillance requirements to provide direct references to Appendix J methodology and terminology.

II. BACKGROUND

Beginning in August 1975, the NRC staff requested licensees to review their containment leakage testing programs and the associated Technical Specifications for compliance with the requirements of Appendix J to 10 CFR Part 50. Recognizing at that time that there were already many operating plants and a number more in advanced stages of design or construction, we requested licensees to propose design modifications and Technical Specification changes and, as necessary, request exemptions to attain conformance with the regulations. The Commonwealth Edison Company had previously responded to our request by letters dated September 26, 1975, September 9, 1976, and April 5, 1977.

As part of that response, the licensee requested a number of exemptions to the provisions of Appendix J. Those requests are under review and are not addressed in this Safety Evaluation.

III. EVALUATION

By letter dated November 30, 1981 the licensee proposed amending the Dresden Nuclear Power Station Units 2 and 3 Technical Specifications (TS) to modify the primary containment integrated leakage testing requirements and schedules to conform with 10 CFR Part 50, Apppendix J requirements. The proposed changes also provided for direct references and use of Appendix J methodology and terminology.

The BWR Standard Technical Specifications, NUREG-0123, Revision 3, served as the basis in assessing the conformance of the licensee's proposed Technical Specification changes to Appendix J requirements. The Standard Technical Specifications, pages 3/4 6-2 through 3/4 6-4, pertaining to primary containment leakage testing requirements (and the associated Bases) are recognized by the staff as an acceptable implementation of the applicable requirements of Appendix J.

We have reviewed the licensee's submittal dated November 30, 1981 and find the licensee's proposed Technical Specification changes to be consistent with the BWR Standard Technical Specifications. Therefore, we conclude that the Technical Specification changes pertaining to containment integrated leakage testing meet the requirements of 10 CFR Part 50, Appendix J, and are acceptable.

IV. ENVIRONMENTAL CONSIDERATIONS

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 environmental 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 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.

V. CONCLUSION

We have concluded, 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.

Date: April 1, 1982

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-237 AND 50-249

COMMONWEALTH EDISON COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO OPERATING LICENSES

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 68 to Provisional Operating License No. DPR-19 and Amendment No. 60 to Facility Operating License No. DPR-25, issued to Commonwealth Edison Company, which revised the Technical Specifications for operation of the Dresden Nuclear Power Station, Units 2 and 3 located in Grundy County, Illinois. The amendments are effective as of the date of issuance.

The changes to the Technical Specifications provide for primary containment integrated leak rate test requirements and schedules consistent with Appendix J to 10 CFR Part 50. The changes also provide for direct references and use of Appendix J methodology and terminology.

The application for the amendments complies 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 51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated November 30, 1981, (2) Amendment No. 68 to License No. DPR-19 and Amendment No. 60 to License No. DPR-25, and (3) 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 at the Morris Public Library, 604 Liberty Street, Morris, Illinois. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 1st day of April 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

D B Vassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing