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Docket No. 50-237,
50-249

NOV 19 1976

Commonwealth Edison Company
ATTN: Mr. R. L. Bolger
Assistant Vice President
Post Office Box 767
Chicago, Illinois 60690

Gentlemen:

In response to your requests dated October 13, and 19, 1976, the Commission has issued the enclosed Amendment Nos. 26 and 25 to Facility Operating License Nos. DPR-19 and DPR-25 for Unit Nos. 2 and 3 of the Dresden Nuclear Power Station, respectively.

These amendments require that the structural integrity of the primary coolant system boundary be maintained at the level required by Section XI of the 1974 Edition of the ASME Boiler and Pressure Vessel Code. This specification supersedes a previous requirement based upon Section III of the 1965 Edition of the ASME Code.

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

Sincerely,

Original signed by
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment No. 26 to License No. DPR-19
2. Amendment No. 25 to License No. DPR-25
3. Safety Evaluation
4. Notice

cc w/enclosures:
See next page

OFFICE ➤	DOR:ORB #2	DOR:ORB #2	OELD	DOR:ORB #2	DOR:AD/OT
SURNAME ➤	RMDiggs	RDSilver:ro	Swanson	DLZiemann	DEisenhut
DATE ➤	11/2/76	11/2/76	11/10/76	11/15/76	11/19/76

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Morris Public Library
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Washington, D. C. 20460

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

Mr. William Waters
Chairman, Board of Supervisors
of Grundy County
Grundy County Courthouse
Morris, Illinois 60450

Department of Public Health
ATTN: Chief, Division of
Radiological Health
535 West Jefferson Street
Springfield, Illinois 62706

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN UNIT NO. 2

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 26
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 October 13, 1976, as supplemented by letter dated October 19, 1976, 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.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 19, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 26

PROVISIONAL OPERATING LICENSE NO. DPR-19

DOCKET NO. 50-237

Replace the following pages of the Technical Specifications contained in Appendix A of the above-indicated license with the attached pages bearing the same numbers, except as otherwise indicated. The changed areas on the pages are reflected by a marginal line.

Remove Pages	Insert Pages
iii - Table of Contents	iii - Table of Contents
91	91
	91a - new
	91a-1 - new
91a	91a-2
98	98
	98a - new

TABLE OF CONTENTS (Cont)

F. Structural Integrity

The structural integrity of the primary system boundary shall be maintained at the level required by the ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components", 1974 Edition, **Summer 1975 Addenda. (ASME) Code Section XI).** Components of the primary system boundary whose inservice examination reveals the absence of flaw indications or flaw indications not in excess of the allowable indication standards of this Code are acceptable for continued service. Plant operation with components which have inservice examination flaw indication(s) in excess of the allowable indication standards of the Code shall be subject to NRC approval.

a. Components whose inservice examination reveals flaw indication(s) in excess of the allowable indication standards of the ASME Code, Section XI, are unacceptable for continued service unless the following requirements are met:

F. STRUCTURAL INTEGRITY

The nondestructive inspections listed in Table 4.6.1 shall be performed as specified in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1971 Edition, Summer 1971 Addenda. The results obtained from compliance with this specification will be evaluated after 5 years and the conclusions of this evaluation will be reviewed with the NRC.

(i) An analysis and evaluation of the detected flaw indication(s) shall be submitted to the NRC that demonstrate that the component structural integrity justifies continued service. The analysis and evaluation shall follow the procedures outlined in Appendix A, "Evaluation of Flaw Indications", of ASME Code, Section XI.

(ii) Prior to the resumption of service, the NRC shall review the analysis and evaluation and either approve resumption of plant operation with the affected component or require that the component be repaired or replaced.

b. For components approved for continued service in accordance with paragraph a. above, reexamination of the area containing the flaw indication(s) shall be conducted during each scheduled successive inservice inspection. An analysis and evaluation shall be submitted to the NRC following each inservice inspection. The analysis and evaluation shall follow the procedures outlined in Appendix A, "Evaluation of Flaw Indications", of ASME Code, Section XI, and shall reference prior analyses submitted to the NRC to the extent applicable. Prior to resumption of service following each inservice inspection, the NRC shall review the analysis and evaluation and either approve resumption of plant operation with the affected component or require that the component be repaired or replaced.

c. Repair or replacement of components, including reexaminations, shall conform with the requirements of the ASME Code, Section XI. In the case of repairs, flaws shall be either removed or repaired to the extent necessary to meet the allowable indication standards specified in ASME Code, Section XI.

G. Jet Pumps

1. Whenever the Reactor is in the Startup/Hot Standby or Run modes, all jet pumps shall be intact and all operating jet pumps shall be operable. If it is determined that a jet pump is inoperable, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

2. Flow indication from each of the twenty jet pumps shall be verified prior to initiation of reactor startup from a cold shutdown condition.

3. The indicated core flow is the sum of the flow indication from each of the twenty jet pumps. If flow indication failure occurs for two or more jet pumps, immediate corrective action shall be taken. If flow indication for all but one jet pump cannot be obtained within 12 hours an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

G. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the Startup/Hot Standby or Run modes, jet pump integrity and operability shall be checked daily by verifying that the following two conditions do not occur simultaneously:

   a. The recirculation pump flow differs by more than 10% from the established speed-flow characteristics.

   b. The indicated total core flow is more than 10% greater than the core flow value derived from established power-core flow relationships.

2. Additionally, when operating with one recirculation pump with the equalizer valves closed, the diffuser to lower plenum differential pressure shall be checked daily, and the differential pressure of any jet pump in the idle loop shall not vary by more than 10% from established patterns.

3.6 LIMITING CONDITION FOR OPERATION	4.6 SURVEILLANCE REQUIREMENT
	3. The baseline data required to evaluate the conditions in Specifications 4.6.G.1 and 4.6.G.2 will be acquired each operating cycle.
H. Recirculation Pump Flow Mismatch	H. Recirculation Pump Flow Mismatch
1. Whenever both recirculation pumps are in operation, pump speeds shall be maintained within 10% of each other when power level is greater than 80% and within 15% of each other when power level is less than 80%.	Recirculation pumps speed shall be checked daily for mismatch.
2. If specification 3.6.H.1 cannot be met, one recirculation pump shall be tripped.	
3. The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service the plant shall be placed in a hot shutdown condition within 24 hours unless the loop is sooner returned to service.	
4. Whenever one pump is operable and the remaining pump is in the tripped position, the operable pump shall be at a speed less than 65% before starting the inoperable pump.	

The performance of reactor coolant leakage detection system will be evaluated during the first five years of station operation and the conclusions of this evaluation will be reported to the NRC.

It is estimated that the main steam line tunnel leakage detection system is capable of detecting of the order of 3000 lb/hr. The system performance will be evaluated during the first five years of plant operation and the conclusions of the evaluation will be reported to the NRC.

E. Safety and Relief Valves — Experience in safety valve operation shows that a testing of 50% of the safety valves per refueling outage is adequate to detect failures or deterioration. The tolerance value is specified in Section III of the ASME Boiler and Pressure Vessel Code as ±1% of design pressure. An analysis has been performed which shows that with all safety valves set 1% higher the reactor coolant pressure safety limit of 1375 psig is not exceeded. Solenoid actuated relief valves are used to avoid activation of the safety valves. In view of the fact that the solenoid activated relief valves are more complicated, it is prudent to test them at each refueling outage. The safety valves are required to be operable above the design pressure (90 psig) at which the core spray subsystems are not designed to deliver full flow.

F. Structural Integrity — A pre-service inspection of the components listed in Table 4.6.1 will be conducted after site erection to assure the system is free of gross defects and as a reference base for later inspections. Prior to operation, the reactor primary system will be free of gross defects. In addition, the facility has been designed such that gross defects should not occur throughout life. The inspection program given in Table 4.6.1 was based on Section XI of the ASME Boiler and Pressure Vessel Code, 1971 Edition, Summer 1971 Addenda, which was followed except where accessibility for inspection was not provided. This edition of the Code is suitable for detecting flaw indications but does not provide adequate guidance for the evaluation of ultrasonic reflectors. The requirement in the 1971 Edition of Section XI that the operator evaluate the reflector to determine the size, shape, and nature can best be satisfied by examination and evaluation of the flaw in accordance with the techniques presented in Appendix A to ASME Section XI in the 1974 Edition, **Summer 1975 addenda. It is the in**tent of this specification to require inservice inspection of the primary system boundary per Table 4.6.1 of this specification and the 1971 Edition of ASME Section XI including the Summer 1971 Addenda and to permit the evaluation of flaws in excess of the acceptance standards of that Edition and Addenda in accordance with the techniques of the 1974 version. Commonwealth Edison Company recognizes the importance of inspection of those areas which are presently not accessible and will study and implement, if practicable, new means to include those areas within the inspection program. This inspection provides further assurance that gross defects are not occurring after the system is in service. This inspection will reveal problem areas should they occur before a leak develops.

The special inspection of the main feed and steam lines is to provide added protection against pipe whip. The GRP I welds are selected on the basis of an analysis that shows these welds are the highest stress welds and that due to their physical location, a break would result in the least interference and maximum energy upon impact with the drywell. These welds are the only ones which offer any significant risk and are therefore inspected four times as often as the other welds within the drywells.

GRP II welds are selected because without regard for the operating stress levels and interfering equipment, they have sufficient theoretical energy to penetrate and would propel the pipe toward the containment. They are therefore included in first inspection. Upon consideration of impact angle, interfering equipment and distance pipe travels, no substantial risk is involved and no extra inspection is needed.

In addition, extensive visual inspection for leaks will be made periodically on critical systems. The inspection program specified encompasses the major areas of the vessel and piping systems within the drywell. The inspection period is based on the observed rate of growth of defects from fatigue studies sponsored by the NRC.

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-249

DRESDEN UNIT NO. 3

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 25
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 October 13, 1976, as supplemented by letter dated October 19, 1976, 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.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 19, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 25

PROVISIONAL OPERATING LICENSE NO. DPR-25

DOCKET NO. 50-249

Replace the following pages of the Technical Specifications contained in Appendix A of the above-indicated license with the attached pages bearing the same numbers, except as otherwise indicated. The changed areas on the pages are reflected by a marginal line.

Remove Pages	Insert Pages
iii - Table of Contents	iii - Table of Contents
91	91
	91a - new
	91a-1 - new
91a	91a-2
98	98
	98a - new

TABLE OF CONTENTS (Cont)

F. Structural Integrity

The structural integrity of the primary system boundary shall be maintained at the level required by the ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components", 1974 Edition, **Summer 1975 Addenda. (ASME) Code Section XI).** Components of the primary system boundary whose inservice examination reveals the absence of flaw indications or flaw indications not in excess of the allowable indication standards of this Code are acceptable for continued service. Plant operation with components which have inservice examination flaw indication(s) in excess of the allowable indication standards of the Code shall be subject to NRC approval.

a. Components whose inservice examination reveals flaw indication(s) in excess of the allowable indication standards of the ASME Code, Section XI, are unacceptable for continued service unless the following requirements are met:

F. STRUCTURAL INTEGRITY

The nondestructive inspections listed in Table 4.6.1 shall be performed as specified in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1971 Edition, Summer 1971 Addenda. The results obtained from compliance with this specification will be evaluated after 5 years and the conclusions of this evaluation will be reviewed with the NRC.

(i) An analysis and evaluation of the detected flaw indication(s) shall be submitted to the NRC that demonstrate that the component structural integrity justifies continued service. The analysis and evaluation shall follow the procedures outlined in Appendix A, "Evaluation of Flaw Indications", of ASME Code, Section XI.

(ii) Prior to the resumption of service, the NRC shall review the analysis and evaluation and either approve resumption of plant operation with the affected component or require that the component be repaired or replaced.

b. For components approved for continued service in accordance with paragraph a. above, reexamination of the area containing the flaw indication(s) shall be conducted during each scheduled successive inservice inspection. An analysis and evaluation shall be submitted to the NRC following each inservice inspection. The analysis and evaluation shall follow the procedures outlined in Appendix A, "Evaluation of Flaw Indications", of ASME Code, Section XI, and shall reference prior analyses submitted to the NRC to the extent applicable. Prior to resumption of service following each inservice inspection, the NRC shall review the analysis and evaluation and either approve resumption of plant operation with the affected component or require that the component be repaired or replaced.

c. Repair or replacement of components, including reexaminations, shall conform with the requirements of the ASME Code, Section XI. In the case of repairs, flaws shall be either removed or repaired to the extent necessary to meet the allowable indication standards specified in ASME Code, Section XI.

G. Jet Pumps

1. Whenever the Reactor is in the Startup/Hot Standby or Run modes, all jet pumps shall be intact and all operating jet pumps shall be operable. If it is determined that a jet pump is inoperable, an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown condition within 24 hours.

2. Flow indication from each of the twenty jet pumps shall be verified prior to initiation of reactor startup from a cold shutdown condition.

3. The indicated core flow is the sum of the flow indication from each of the twenty jet pumps. If flow indication failure occurs for two or more jet pumps, immediate corrective action shall be taken. If flow indication for all but one jet pump cannot be obtained within 12 hours an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

G. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the Startup/Hot Standby or Run modes, jet pump integrity and operability shall be checked daily by verifying that the following two conditions do not occur simultaneously:

   a. The recirculation pump flow differs by more than 10% from the established speed-flow characteristics.

   b. The indicated total core flow is more than 10% greater than the core flow value derived from established power-core flow relationships.

2. Additionally, when operating with one recirculation pump with the equalizer valves closed, the diffuser to lower plenum differential pressure shall be checked daily, and the differential pressure of any jet pump in the idle loop shall not vary by more than 10% from established patterns.

3.6 LIMITING CONDITION FOR OPERATION

H. Recirculation Pump Flow Mismatch

1. Whenever both recirculation pumps are in operation, pump speeds shall be maintained within 10% of each other when power level is greater than 80% and within 15% of each other when power level is less than 80%.

2. If specification 3.6.H.1 cannot be met, one recirculation pump shall be tripped.

3. The reactor shall not be operated with one recirculation loop out of service for more than 24 hours. With the reactor operating, if one recirculation loop is out of service the plant shall be placed in a hot shutdown condition within 24 hours unless the loop is sooner returned to service.

4. Whenever one pump is operable and the remaining pump is in the tripped position, the operable pump shall be at a speed less than 65% before starting the inoperable pump.

4.6 SURVEILLANCE REQUIREMENT

3. The baseline data required to evaluate the conditions in Specifications 4.6.G.1 and 4.6.G.2 will be acquired each operating cycle.

H. Recirculation Pump Flow Mismatch

Recirculation pumps speed shall be checked daily for mismatch.

The performance of reactor coolant leakage detection system will be evaluated during the first five years of station operation and the conclusions of this evaluation will be reported to the NRC.

It is estimated that the main steam line tunnel leakage detection system is capable of detecting of the order of 3000 lb/hr. The system performance will be evaluated during the first five years of plant operation and the conclusions of the evaluation will be reported to the NRC.

E. Safety and Relief Valves — Experience in safety valve operation shows that a testing of 50% of the safety valves per refueling outage is adequate to detect failures or deterioration. The tolerance value is specified in Section III of the ASME Boiler and Pressure Vessel Code as ±1% of design pressure. An analysis has been performed which shows that with all safety valves set 1% higher the reactor coolant pressure safety limit of 1375 psig is not exceeded. Solenoid actuated relief valves are used to avoid activation of the safety valves. In view of the fact that the solenoid activated relief valves are more complicated, it is prudent to test them at each refueling outage. The safety valves are required to be operable above the design pressure (90 psig) at which the core spray subsystems are not designed to deliver full flow.

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GRP II welds are selected because without regard for the operating stress levels and interfering equipment, they have sufficient theoretical energy to penetrate and would propel the pipe toward the containment. They are therefore included in first inspection. Upon consideration of impact angle, interfering equipment and distance pipe travels, no substantial risk is involved and no extra inspection is needed.

In addition, extensive visual inspection for leaks will be made periodically on critical systems. The inspection program specified encompasses the major areas of the vessel and piping systems within the drywell. The inspection period is based on the observed rate of growth of defects from fatigue studies sponsored by the NRC.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 26 AND 25 TO

FACILITY LICENSE NOS. DPR-19 AND DPR-25

COMMONWEALTH EDISON COMPANY

DRESDEN UNIT NOS. 2 AND 3

DOCKET NOS. 50-237 AND 50-249

INTRODUCTION

By letter dated October 13, 1976, as supplemented by letter dated October 19, 1976, the Commonwealth Edison Company requested amendments to Facility License Nos. DPR-19 and DPR-25 for the Dresden Nuclear Power Station Unit Nos. 2 and 3. The proposed request involves revisions to the Technical Specifications with regard to updating the ASME Code requirements to which Commonwealth Edison maintains the structural integrity of the primary coolant system boundary.

DISCUSSION

The Technical Specifications for Dresden Unit Nos. 2 and 3 currently require that the structural integrity of the primary coolant system boundary be maintained at the level required by the original acceptance standards. For Dresden Unit Nos. 2 and 3, these original acceptance standards are contained in Section III of the 1965 Edition of the ASME Code. Subsequent to the issuance of the operating licenses for Dresden 2 and 3, the Technical Specifications were revised to require that Commonwealth Edison conduct an inservice inspection program at these units that is in accordance with the provisions of Section XI of the ASME Code, "Rules for Inservice Inspection of Nuclear Power Plant Components." Section XI also contains acceptance standards for the structural integrity of the components examined. The proposed technical specification revisions require that the acceptance standards prescribed in Section XI of the ASME Code be applied to the examinations made in accordance with that code.

EVALUATION

The NRC staff has reviewed Commonwealth Edison's proposed revision to the Dresden Unit Nos. 2 and 3 Technical Specifications which would require the primary coolant system structural integrity to be maintained at the level required by Section XI of the 1974 Edition of the ASME Boiler and Pressure Vessel Code, "Rules for Inservice Inspection of Nuclear Power Plant Components."

We conclude that because Section XI of the ASME Code contains the acceptance standards for evaluating the results of the inservice inspections required by the Dresden Unit Nos. 2 and 3 Technical Specifications, it should be specified as the acceptance standard for the structural integrity of these plants. This conclusion is based upon the following:

1. The original acceptance criteria of Section III of the ASME Code were fabrication criteria and were not intended to be used as inservice acceptance criteria.

2. The acceptance criteria of Section XI of the ASME Code have been explicitly developed for use as acceptance criteria for inservice inspections conducted on nuclear power plant components.

3. The use of the acceptance criteria of Section XI of the ASME Code is consistent with the Commission's current regulations contained in 10 CFR 50.55a. The use of Section XI acceptance criteria is also consistent with the Commission's Regulatory position as expressed in its Standard Technical Specifications for Boiling Water Reactors. The application of the acceptance criteria of Section XI of the ASME Code to Dresden Unit Nos. 2 and 3 does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin.

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 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.

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: November 19, 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-237 AND 50-249

COMMONWEALTH EDISON COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 26 and 25 to Facility Operating License Nos. DPR-19 and DPR-25, issued to Commonwealth Edison Company (the licensee), which revised Technical Specifications for operation of the Dresden Station Unit Nos. 2 and 3 (the facilities) located in Grundy County, Illinois. The amendments are effective as of their date of issuance.

These license amendments revised the Technical Specifications for the facilities to require that the structural integrity of the primary coolant system boundary be maintained at the level required by Section XI of the 1974 Edition of the ASME Boiler and Pressure Vessel Code. This specification supersedes a previous requirement that was based upon Section III of the 1965 Edition of the ASME Code.

The application, as supplemented, 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 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 October 13, 1976, and supplement thereto dated October 19, 1976, (2) Amendment Nos. 26 and 25 to License Nos. DPR-19 and 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 60451. 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 Operating Reactors.

Dated at Bethesda, Maryland, this 19th day of November, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors