March 11, 1977

Docket Nos. 50-237
and 50-249

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

Gentlemen:

In response to your request dated January 27, 1977, the Commission has issued the enclosed Amendment Nos. 28 and 27 to Facility Operating License Nos. DPR-19 and DPR-25 for Unit Nos. 2 and 3 of the Dresden Nuclear Power Station, respectively.

The amendments consist of changes in the Technical Specifications related to coupling of control rods to their drives and explicitly authorizes attempts to recouple a rod to its drive at power levels above 20% rated power. During our reivew we found that certain changes to your proposal were necessary. Your staff has agreed to these changes and they have been incorporated.

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

Since February 1973 nine control rods at Dresden Unit 2 have uncoupled from their drives. In a letter of July 17, 1973 (WPW Ltr. #511-73) you indicated that the 1973 occurrences may have been the result of improper installation of control rod drive inner filters during the Spring 1973 refueling outage. In addition, you stated that "Additionally, when any drive is removed for scheduled maintenance, we will insure that the inner filter is properly installed by following the correct procedures." In 1974, four rods at Dresden 2 uncoupled from drives that had inner filters installed in 1972. In a letter of May 16, 1975 (BBS Ltr. #311-75) you concluded that the uncouplings to date were the result of retainer spring damage and/or improper assembly techniques during the Spring 1972 refueling outage. You also stated that all remaining CRD's overhauled in 1972 had been inspected, overhauled and reinstalled into the reactor during the Winter 1974 refueling outage. In December 1976 two additional drives uncoupled from their drives. Our understanding is that these drives had been overhauled and reinstalled in the reactor during the Winter 1974 refueling outage. Therefore, the problem cannot be attributed entirely to improper installation during the 1972 outage, and you apparently have not taken adequate steps to insure that the inner filter is properly installed.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

Although the uncouplings which you are experiencing do not present a threat to public health and safety or involve a significant hazards consideration, we do not consider it prudent to continue to accept control rod uncoupling at Dresden Unit 2. We therefore request that you develop a program for implementation during the forthcoming refueling outage to eliminate the uncoupling occurrences. We further request that your proposed program be submitted for our review and approval within 60 days of the date of this letter.

Sincerely,

Original Signed by:
Dennis L. Ziemann

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

Enclosures:
1. Amendment No. 28 to
   License No. DPR-19
2. Amendment No. 27 to
   License No. DPR-25
3. Safety Evaluation
4. Notice

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JRBuchanan

I discussed the addition of the last paragraph of the transmittal letter with Dan Swanson of OELD and Frank Coffman of OT. Neither had any objection to the paragraph and neither saw any reason to have this letter and amendment rerouted for formal concurrences. R. Silver

SEE PREVIOUS YELLOW FOR CONCURRENCES

OFFICE	DOR:ORB #2	DOR:ORB #2	DOR:RSB/OT	DOR:AD/OT	OELD	DOR:ORB #2
SURNAME	RMDiggs	RDSilver:rm	RBaer	DEisenhut		DLZiemann
DATE	3/ /77	3/11/77	3/ /77	3/ /77	3/ /77	3/ /77

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DRoss

Docket Nos. 50-237
and 50-249

MAR 11 1977

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

Gentlemen:

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A copy of the related Safety Evaluation and 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. to License No. DPR-19
2. Amendment No. to License No. DPR-25
3. Safety Evaluation
4. Notice

Informed CECo & OIE on 3/11/77 R. Silver

cc w/enclosures: See next page

OFFICE ➤	ORB#2:DOR	ORB#2:DOR	C-RSB-OT&DOR	AD-OT:DOR	OELD	C-ORB#2:DOR
SURNAME ➤	RDiggs	RSilver:rm	RBaer	DEisenhut	D Swanson	DZiemann
DATE ➤	3/2/77	3/1/77	3/3/77	3/4/77	3/11/77	3/1/77

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

March 11, 1977

Docket Nos. 50-237
and 50-249

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

Gentlemen:

In response to your request dated January 27, 1977, the Commission has issued the enclosed Amendment Nos. 28 and 27 to Facility Operating License Nos. DPR-19 and DPR-25 for Unit Nos. 2 and 3 of the Dresden Nuclear Power Station, respectively.

The amendments consist of changes in the Technical Specifications related to coupling of control rods to their drives and explicitly authorizes attempts to recouple a rod to its drive at power levels above 20% rated power. During our reivew we found that certain changes to your proposal were necessary. Your staff has agreed to these changes and they have been incorporated.

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Since February 1973 nine control rods at Dresden Unit 2 have uncoupled from their drives. In a letter of July 17, 1973 (WPW Ltr. #511-73) you indicated that the 1973 occurrences may have been the result of improper installation of control rod drive inner filters during the Spring 1973 refueling outage. In addition, you stated that "Additionally, when any drive is removed for scheduled maintenance, we will insure that the inner filter is properly installed by following the correct procedures." In 1974, four rods at Dresden 2 uncoupled from drives that had inner filters installed in 1972. In a letter of May 16, 1975 (BBS Ltr. #311-75) you concluded that the uncouplings to date were the result of retainer spring damage and/or improper assembly techniques during the Spring 1972 refueling outage. You also stated that all remaining CRD's overhauled in 1972 had been inspected, overhauled and reinstalled into the reactor during the Winter 1974 refueling outage. In December 1976 two additional drives uncoupled from their drives. Our understanding is that these drives had been overhauled and reinstalled in the reactor during the Winter 1974 refueling outage. Therefore, the problem cannot be attributed entirely to improper installation during the 1972 outage, and you apparently have not taken adequate steps to insure that the inner filter is properly installed.

Although the uncouplings which you are experiencing do not present a threat to public health and safety or involve a significant hazards consideration, we do not consider it prudent to continue to accept control rod uncoupling at Dresden Unit 2. We therefore request that you develop a program for implementation during the forthcoming refueling outage to eliminate the uncoupling occurrences. We further request that your proposed program be submitted for our review and approval within 60 days of the date of this letter.

Sincerely,

Dennis L. Ziemann

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

Enclosures:
1. Amendment No. 28 to
   License No. DPR-19
2. Amendment No. 27 to
   License No. DPR-25
3. Safety Evaluation
4. Notice

cc w/enclosures:
See next page

cc w/enclosures:
Mr. John W. Rowe
Isham, Lincoln & Beale
Counselors at Law
One First National Plaza, 42nd Floor
Chicago, Illinois 60603

Anthony Z. Roisman, Esquire
Roisman, Kessler and Cashdan
1025 15th Street, N. W., 5th Floor
Washington, D. C. 20005

Morris Public Library
604 Liberty Street
Morris, Illinois 60451

Chief, Energy Systems
Analyses Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 645, East Tower
401 M Street, S. W.
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

cc w/enclosures and cy of CECo
filing dtd. 1/27/77:
Department of Public Health
ATTN: Chief, Division of
Radiological Health
535 West Jefferson
Springfield, Illinois 62706

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION UNIT NO. 2

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 28
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 January 27, 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-19 is hereby amended to read as follows:

"B. <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 28 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications."

3. The 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: March 11, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 28

PROVISIONAL OPERATING LICENSE NO. DPR-19

DOCKET NO. 50-237

Replace the following existing pages of the Technical Specifications with the attached revised pages. Changed areas on the revised pages are shown by a marginal line.

REMOVE PAGES	INSERT PAGES
56	56
62	62

3.3 LIMITING CONDITION FOR OPERATION

B. Control Rods

1. **All control rods shall be coupled to their drive mechanisms when the mode switch is in "Startup" or "Run". With** a control rod not coupled to its associated drive mechanism, operation may continue provided:

   a. Below 20% power, the rod shall be declared inoperable, full inserted, and the directional control valves electrically disarmed until recoupling can be attempted at all-rods-in or at power levels above 20 percent power.

   b. Above 20% power, recoupling is being attempted in accordance with an established procedure or the rod shall be declared inoperable, fully inserted and the directional control valves electrically disarmed.

2. **The control rod drive housing support system shall be in place during reactor power operation and when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.**

4.3 SURVEILLANCE REQUIREMENT

B. Control Rods

1. Coupling Integrity

   a. The coupling integrity of each control rod shall be demonstrated by withdrawing each control rod to the fully withdrawn position and verifying that the rod does not go to the overtravel position;

      (1) Prior to reactor criticality after completing alteration of the reactor core,

      (2) Anytime the control rod is withdrawn to the "Full out" position in subsequent operation, and

      (3) For specifically affected individual control rods following maintenance on or modification to the control rod or rod drive system which could affect the rod drive coupling integrity.

   b. Verify that the control rod is following the drive by observing a response in the nuclear instrumentation each time a rod is moved. When no response is discernible, the response should be verified when the reactor is operating at power levels above 20%.

2. **The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.**

indicative of a generic control rod drive problem and the reactor will be shutdown. Also if damage within the control rod drive mechanism and in particular, cracks in drive internal housings, cannot be ruled out, then a generic problem affecting a number of drives cannot be ruled out. Circumferential cracks resulting from stress assisted intergranular corrosion have occurred in the collet housing of drives at several BWRs. This type of cracking could occur in a number of drives and if the cracks propagated until severance of the collet housing occurred, scram could be prevented in the affected rods. Limiting the period of operation with a potentially severed collet housing and requiring increased surveillance after detecting one stuck rod will assure that the reactor will not be operated with a large number of rods with failed collet housings.

B. Control Rod Withdrawal

1. Control rod dropout accidents as discussed in the SAR can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod dropout accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides a verification that the rod is following its drive. Absence of such response to drive movement would provide cause for suspecting a rod to be uncoupled and stuck. Restricting recoupling verifications to power levels above 20% provides assurance that a rod drop during a recoupling verification would not result in a rod drop accident.

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in Section 6.6.1 of the SAR, and the design evaluation is given in Section 6.6.3. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. Additionally, the support is not required if all control rods are fully inserted and if an adequate shutdown margin with one control rod withdrawn has been demonstrated since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.

3. Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn could not be worth enough to cause the core to be more than 0.013 delta K supercritical if they were to drop out of the core in the manner defined for the Rod Drop Accident.(3) These sequences are developed prior to initial operation of the unit following any refueling outage and the requirement that an operator follow these sequences is backed up by the operation of the RWM. This 0.013 delta K limit, together with the integral rod velocity limiters and the action of the control rod drive system, limit potential reactivity insertion such that the results of a control rod drop accident will not exceed a maximum fuel energy content of 280 cal/gm. The peak fuel enthalpy of 280 cal/gm is below the energy content at which rapid fuel dispersal and primary system damage have been found to occur based on experimental data as is discussed in Reference 1.

The analysis of the control rod drop accident was originally presented in Sections 7.9.3, 14.2.1.2 and 14.2.1.4 of the Safety Analysis Report. Improvements in analytical capability have allowed a more refined analysis of the control rod drop accident.

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-249

DRESDEN NUCLEAR POWER STATION UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 27
License No. DPR-25

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

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated January 27, 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-25 is hereby amended to read as follows:

"B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 27, 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 L. Ziemann

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

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 11, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 27

FACILITY OPERATING LICENSE NO. DPR-25

DOCKET NO. 50-249

Replace the following existing pages of the Technical Specifications with the attached revised pages. Changed areas on the revised pages are shown by a marginal line.

REMOVE PAGES	INSERT PAGES
56	56
62	62

3.3 LIMITING CONDITION FOR OPERATION

B. Control Rods

1. All control rods shall be coupled to their drive mechanisms when the mode switch is in "Startup" or "Run". With a control rod not coupled to its associated drive mechanism, operation may continue provided:

   a. Below 20% power, the rod shall be declared inoperable, full inserted, and the directional control valves electrically disarmed until recoupling can be attempted at all-rods-in or at power levels above 20 percent power.

   b. Above 20% power, recoupling is being attempted in accordance with an established procedure or the rod shall be declared inoperable, fully inserted and the directional control valves electrically disarmed.

2. The control rod drive housing support system shall be in place during reactor power operation and when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.

4.3 SURVEILLANCE REQUIREMENT

B. Control Rods

1. Coupling Integrity

   a. The coupling integrity of each control rod shall be demonstrated by withdrawing each control rod to the fully withdrawn position and verifying that the rod does not go to the overtravel position;

      (1) Prior to reactor criticality after completing alteration of the reactor core,

      (2) Anytime the control rod is withdrawn to the "Full out" position in subsequent operation, and

      (3) For specifically affected individual control rods following maintenance on or modification to the control rod or rod drive system which could affect the rod drive coupling integrity.

   b. Verify that the control rod is following the drive by observing a response in the nuclear instrumentation each time a rod is moved. When no response is discernible, the response should be verified when the reactor is operating at power levels above 20%.

2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.

indicative of a generic control rod drive problem and the reactor will be shutdown. Also if damage within the control rod drive mechanism and in particular, cracks in drive internal housings, cannot be ruled out, then a generic problem affecting a number of drives cannot be ruled out. Circumferential cracks resulting from stress assisted intergranular corrosion have occurred in the collet housing of drives at several BWRs. This type of cracking could occur in a number of drives and if the cracks propagated until severance of the collet housing occurred, scram could be prevented in the affected rods. Limiting the period of operation with a potentially severed collet housing and requiring increased surveillance after detecting one stuck rod will assure that the reactor will not be operated with a large number of rods with failed collet housings.

E. Control Rod Withdrawal

1. Control rod dropout accidents as discussed in the SAR can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod dropout accident is eliminated. The overtravel position feature provides a positive check as only uncoupled drives may reach this position. Neutron instrumentation response to rod movement provides a verification that the rod is following its drive. Absence of such response to drive movement would provide cause for suspecting a rod to be uncoupled and stuck. Restricting recoupling verifications to power levels above 20% provides assurance that a rod drop during a recoupling verification would not result in a rod drop accident.

2. The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in Section 6.6.1 of the SAR, and the design evaluation is given in Section 6.6.3. This support is not required if the reactor coolant system is at atmospheric pressure since there would then be no driving force to rapidly eject a drive housing. Additionally, the support is not required if all control rods are fully inserted and if an adequate shutdown margin with one control rod withdrawn has been demonstrated since the reactor would remain subcritical even in the event of complete ejection of the strongest control rod.

3. Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn could not be worth enough to cause the core to be more than 0.013 delta K supercritical if they were to drop out of the core in the manner defined for the Rod Drop Accident.[3] These sequences are developed prior to initial operation of the unit following any refueling outage and the requirement that an operator follow these sequences is backed up by the operation of the RWM. This 0.013 delta K limit, together with the integral rod velocity limiters and the action of the control rod drive system, limit potential reactivity insertion such that the results of a control rod drop accident will not exceed a maximum fuel energy content of 280 cal/gm. The peak fuel enthalpy of 280 cal/gm is below the energy content at which rapid fuel dispersal and primary system damage have been found to occur based on experimental data as is discussed in Reference 1.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 28 AND 27 TO

FACILITY LICENSE NOS. DPR-19 AND DPR-25

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION UNIT NOS. 2 AND 3

DOCKET NOS. 50-237 AND 50-249

INTRODUCTION

By letter dated January 27, 1977, Commonwealth Edison Company (CE) requested an amendment to Operating Licenses DPR-19 and DPR-25 for the Dresden Nuclear Power Station Unit Nos. 2 and 3. The amendment request would revise the Technical Specifications to allow re-coupling of control rods to their drives in the event uncoupling is observed. During our review of the proposed amendment we found that certain modifications were necessary. Commonwealth Edison representatives have agreed to these changes and they have been incorporated into the proposed Technical Specifications.

DISCUSSION AND EVALUATION

Technical Specification 3.3.B.1 requires that "Each control rod shall be coupled to its drive or completely inserted and the control rod directional or control valves disarmed electrically." If this requirement is not met the Technical Specifications require that an orderly shutdown be initiated and the reactor shall be in cold shutdown condition within 24 hours. The specification is not clear as to whether a re-coupling verification could be attempted following an indication of uncoupling. CE has proposed a change to explicitly allow attempts to recouple a rod to its drive at power levels above 20 percent power.

The safety consideration related to rod coupling verifications is that the verification procedures should not create an opportunity for a control rod drop accident which would cause rapid fuel dispersal. An opportunity for a control rod drop would be created if the control rod drive is lowered away from an uncoupled control rod which is stuck in a partially or fully inserted position. If the stuck rod then suddenly dropped, a step reactivity insertion would occur. This type of accident would not cause fuel damage if the reactor power was above 20% of rated level. Above this power level even a single operator error (withdrawal of a control rod out of specified sequence) cannot result in control rod reactivity worths large enough to cause a peak fuel enthalpy of 280 calories/gram should a control rod drop accident occur. The peak fuel enthalpy of 280 calories/gram is below the energy content at which rapid fuel dispersal and primary system damage would occur. Therefore attempts to recouple at a power level above 20% of rated power would not increase the likelihood of a damaging rod drop accident. The explicit authority to attempt recoupling is consistent with our position for boiling water reactors which have been issued Standard Technical Specifications. Based on these considerations, attempts to recouple control rods under the specified conditions would not increase the opportunity for creating a damaging accident, would not affect safety margins, and would be consistent with our license requirements on other boiling water reactors.

In addition to modifying the recoupling specifications, we have modified the control rod coupling integrity surveillance requirements. Current Technical Specifications only require coupling verification when rod is fully withdrawn the first time subsequent to a refueling outage or after maintenance. We have modified this specification by requiring that coupling verification be performed (a) prior to reactor criticality after completing core alterations which could have affected control rod coupling, (b) anytime a control rod is withdrawn to the "full out" position, and (c) following maintenance on or modification to a control rod or rod drive which could affect rod drive coupling integrity. The first requirement provides additional assurance that a rod drop accident could not occur at the power level range between criticality and 20% of rated power level. In this power range, a rod drop accident could cause fuel damage if control rods were being withdrawn or inserted out of proper sequence. The second requirement provides additional assurance that any uncoupling which occurred would be discovered quickly. Experience at Dresden 2 indicates that the most likely cause of uncoupling is a dislodged inner filter in the control rod drive. The displaced filter could cause uncoupling only when the drive is fully withdrawn. If a coupling verification is performed each time the rod is fully withdrawn, an uncoupling, caused by a displaced filter, would be detected immediately. The third requirement adds assurance that a maintenance has not resulted in an uncoupling. We have discussed these modifications with Commonwealth Edison representatives and they find these changes to be acceptable. The added requirements also are consistent with our position for boiling water reactors which have been issued Standard Technical Specifications.

The technical specification change also makes an editorial change to the limiting condition for operation associated with coupling to clarify when the limiting condition for operation applies.

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: March 11, 1977

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. 28 and 27 to Facility Operating License Nos. DPR-19 and DPR-25, respectively, issued to the Commonwealth Edison Company (the licensee), which revised Technical Specifications for operation of the Dresden Nuclear Power Station Units 2 and 3 (the facilities), located in Grundy County, Illinois. The amendments are effective as of their date of issuance.

These amendments revised the Technical Specifications related to coupling of control rods to their drives and explicitly authorizes attempts to recouple a rod to its drive at power levels above 20% of rated power.

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 these 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 the amendments dated January 27, 1977, (2) Amendment No. 28 to License No. DPR-19, (3) Amendment No. 27 to License No. DPR-25, and (4) the Commission's concurrently issued 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 single copy of items (2) through (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 11th day of March, 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann

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