November 3, 1983

Docket No. 50-237/249
LS05-83-11-014

Mr. Dennis L. Farrar
Director of Nuclear Licensing
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Farrar:

SUBJECT: VERIFICATION OF NUCLEAR RESPONSE TO CONTROL ROD DRIVE MOTION

Dresden Nuclear Power Station, Unit Nos. 2 and 3

The Commission has issued the enclosed Amendment No. 77 to Provisional Operating License No. DPR-19 and Amendment No. 68 to Facility Operating License No. DPR-25 for the Dresden Nuclear Power Station, Units 2 and 3, respectively. The amendments consist of changes to the Technical Specifications in response to your application dated May 24, 1978 as supplemented by letters dated July 15, 1981 and May 2, 1983.

The amendments authorize changes to the Technical Specifications which limit the requirement to verify nuclear response to control rod drive (CRD) motion only for those CRDs that have previously experienced uncoupling. The change clarifies the intent of surveillance requirement 4.3.B.1.b and makes it consistent with the description and analysis in the Final Safety Analysis Report.

A Notice of Consideration of Issuance of Amendment to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested action was published in the Federal Register on September 21, 1983 (48 FR 43131). No request for hearing was received and no comments were received.

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A copy of our related Safety Evaluation is also enclosed. This action will appear in the Commission's Monthly Notice publication in the Federal Register.

Sincerely,

Original signed by

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

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

cc w/enclosures:
See next page

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

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

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

James G. Keppler, Regional Administrator
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799 Roosevelt Road
Glen Ellyn, Illinois 60137

Mr. Gary N. Wright, Manager
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Illinois Department of Nuclear Safety
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION, UNIT NO. 2

DOCKET NO. 50-237

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 77
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 May 24, 1978, as supplemented by letters dated July 15, 1981 and May 3, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 3.B of Provisional Operating License No. DPR-19 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A as revised through Amendment No. 77, 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: November 3, 1983

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

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION, UNIT NO. 3

DOCKET NO. 50-249

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 68
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 May 24, 1978, as supplemented by letters dated July 15, 1981 and May 3, 1983, 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. 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: November 3, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 77*

PROVISIONAL OPERATING LICENSE NO. DPR-19 AND

AMENDMENT NO. 68* TO FACILITY OPERATING LICENSE NO. DPR-25

DOCKET NOS. 50-237/249

Replace page 56 of the Appendix A Technical Specifications with the enclosed page 56. This revised page contains the captioned amendment number and a vertical line indicating the change.

*During the issuance of License Amendment No. 76 to DPR-19 and License Amendment No. 67 to DPR-25 a date and the amendment nos. were inadvertently omitted from pages 91b and 99b. Therefore, corrected pages are attached hereto.

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 percent 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. Normal operating practice is to observe the expected response of the nuclear instrumentation to verify that the control rod is following its drive each time that control rod is withdrawn. For control rod drives that have experienced uncoupling and no response is discernable on the nuclear instrumentation, the response should be verified when the reactor is operating at power levels above 20 percent.

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

3.6 LIMITING CONDITION FOR OPERATION

I. Snubbers (Shock Suppressors)

1. During all modes of operation except cold shutdown and refuel, all safety related snubbers limited in Table 3.6.1a and 3.6.1b shall be operable except as noted in Specification 3.6.1.2 through 3.6.1.4.

2. From and after the time a snubber is determined to be inoperable, continued reactor operation is permissible only during the succeeding 72 hours unless the snubber is sooner made operable or replaced. Torus Ring Header snubbers may be inoperable in either of the following configurations until January 19, 1984, to facilitate the installation of the Mark I torus attached piping modifications.

   Configuration A: Every other existing snubber pair (up to 3 pairs) on the ECCS header, or

   Configuration B: One existing snubber from each of the 6 existing snubber pairs on the ECCS header.

3. If the requirements of 3.6.I.1 and 3.6.I.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in cold shutdown or refuel condition within 36 hours.

4. If a snubber is determined to be inoperable while the reactor is in the cold shutdown or refuel mode, the snubber shall be made operable or replaced prior to reactor startup. This requirement does not apply to Torus Ring Header snubbers for the period identified in paragraph 3.6.I.2 above.

5. Snubbers may be added to safety related systems without prior license amendment to Tables 3.6.1a and/or 3.6.1b provided that a revision to Tables 3.6.1a and/or 3.6.1b is included with the next license amendment request.

4.6 SURVEILLANCE REQUIREMENT

I. Snubbers (Shock Suppressors)

The following surveillance requirements apply to safety related snubbers listed in Tables 3.6.1a and 3.6.1b.

1. Visual Inspection

An independent visual inspection shall be performed on the safety related hydraulic and mechanical snubbers contained in Tables 3.6. and 3.6.1b in accordance with the below schedu

a. All hydraulic snubbers whose seal material has been demonstrated by operating experience, lab testing or analysis to be compatible with the operating environment shall be visually inspected. This inspection shall include, but not necessarily be limited to, inspection of the hydraulic fluid reservoir, fluid connections, and linkage connection to the piping and ancho to verify snubber operability.

b. All mechanical snubbers shall be visually inspected. This inspection shall consist of, but not necessarily be limited to, inspection of the snubber and attachments to the piping and anchor for indications of damage or impaired operability.

No. of Snubbers Found Inoperable During Inspection Interval	Next Required Inspection Interval
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3, 4	124 days ± 25%
5, 6, 7	62 days ± 25%
≥8	31 days ± 25%

The inspection frequency is based upon maintaining a constant level of snubber protection. Thus, the required inspection interval varies inversely with the observed snubber failures. The number of inoperable snubbers found during a required inspection determines the time interval for the next required inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original require time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

To further increase the assurance of snubber reliability, functional tests will be performed once each refueling cycle. A representative sample of 10% of the safety-related snubbers will be functionally tested. Observed failures on these samples will require testing of additional units.

Hydraulic snubbers and mechanical snubbers may each be treated as different entities for the above surveillance programs.

Hydraulic snubber testing will include stroking of the snubbers to verify piston movement, lock-up, and bleed. Functional testing of the mechanical snubbers will consist of verification that the force that initiates free movement of the snubber in either tension or compression is less than the maximum breakaway friction force. The remaining portion of the functional test consisting of verification that the activation (restraining action) is achieved within the specified range of acceleration in both tension and compression will not be done. This is due to the lack of competitive marketable test fixtures available for station use. Therefore, until such time as test fixtures become available, only part (i) of the test will be performed; part (ii) will not be done.

When the cause of rejection of the snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration.

Monitoring of snubber service life shall consist of the existing station record systems, including the central filing system, maintenance files, safety-related work packages, and snubber inspection records. The record retention programs employed at the station shall allow station personnel to maintain snubber integrity. The service life for hydraulic snubbers is 10 years. The hydraulic snubbers existing locations do not impose undue safety implications on the piping and components because they are not exposed to excesses in environmental conditions. The service life for mechanical snubbers is 40 years, lifetime of the plant. The mechanical snubbers are installed in areas of harsh environmental conditions because of their dependability over hydraulic snubbers in these areas. All snubber installations have been thoroughly engineere providing the necessary safety requirements. Evaluations of all snubber locations and environmental conditions justify the above conservative snubber service lives.

A re-analysis of the ring header design based upon acceleration response spectra derived from the original suction header analysis report demonstrates that for normal operation plug seismic, neither the header nor the torus penetration are over-stressed with all snubbers inoperable. The limitation of a maximum of 3 pairs or 1 snubber from each pair inoperable out of 6 pairs is considered conservative. Since the analysis shows that the plant can operate safely indefinitely with no snubbers on the ring header the limitation on operation and startup with inoperable snubbers until January 19, 1984 is justified. This time frame is adequate to allow completion of the Mark I torus attached piping modification.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 77 TO PROVISIONAL OPERATING LICENSE NO. DPR-19

AND AMENDMENT NO. 68 TO FACILITY OPERATING LICENSE NO. DPR-25

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION, UNIT NOS. 2 AND 3

DOCKET NOS. 50-237 AND 50-249

1.0 INTRODUCTION

By letter dated May 24, 1978, as supplemented July 15, 1981, and May 2, 1983, Commonwealth Edison Company (CECo) (the licensee) proposed amendments to Appendix A of Operating License Nos. DPR-19 and DPR-25. The subject change involves Section 4.3.B.1.b of the Technical Specifications for Dresden Unit Nos. 2 and 3. The licensee has proposed to amend Section 4.3.B.1.b, Surveillance Requirements for Control Rod Coupling Integrity, to modify the wording for better understanding.

A Notice of Consideration of Issuance of Amendment to License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing related to the requested action was published in the Federal Register on September 21, 1983 (48 FR 43131). No request for hearing was received and no comments were received.

2.0 EVALUATION

Because of an early history of occasional control rod uncoupling in the Dresden reactors, Technical Specification surveillance for uncoupling verifications in addition to those now required in the Standard Technical Specifications for GeneraL Electric Boiling Water Reactors, NUREG-0123, were included as specification 4.3.B.1.b. This required confirmation of coupling using nuclear instrument response during a rod notch withdrawal. When no instrument response was discernable at lower power, the response should be verified when the reactor is operating at power levels about 20%. The intent of the specification was to provide a general check for all control rods but specifically for those rods with an uncoupling history.

The existing Technical Specifications 4.3.B.1.b wording implies that all rods be reverified at power levels above 20%.

To clarify the intent of surveillance requirement 4.3.B.1.b, CECo proposed a wording change in a letter dated May 24, 1978. In a letter dated May 2, 1983, the wording was modified to further clarify the intent and consistency with the description and analysis in the Final Safety Analysis Report (FSAR).

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The Final Safety Analysis Report (FSAR) Section 3.5.4, Surveillance and Testing for Control Rods, states in part "...During reactor operation individual control rod drive mechanisms can be actuated to demonstrate functional performance. Each time a control rod is withdrawn a notch, the operator will observe the in-core monitors' indications to verify that the control rod is following the drive mechanism.

When the operator withdraws a control rod full out of the core, he tests the coupling integrity by trying to withdraw the rod drive mechanism to the overtravel position. Failure of the drive to overtravel demonstrates rod to drive coupling integrity."

FSAR, Section 6.5.1, Design Basis for Control Rod Velocity Limiters, states in part "...The purpose of the control rod velocity limiter is to reduce the consequences in the event a high-worth control rod became detached from its rod drive and dropped out of the reactor core."

FSAR Section 14.2.1, Control Rod Drop Accident, shows that the analysis is based upon a fully inserted control rod assumed to fall out of the core after becoming disconnected from its drive and after the drive has been removed to the fully withdrawn position. In order to assure that the control rod remains connected to its drive, and in the interest of good operating practices, the licensee's proposed change to Technical Specification 4.3.B.1.b, reaffirms the FSAR. Further, when the nuclear instrumentation does not provide evidence of the control rod movement, e.g., during a startup, the proof of coupling integrity for rods with uncoupling history will be conducted at a power level in excess of 20% where local power range monitors will give the necessary indication.

In addition, the licensee submittal of July 15, 1981 references General Electric Service Information Letter, SIL #52, Supplement 2, July 31, 1974, which shows that improper installation of the control rod drive inner filter had resulted in causing control rods to become uncoupled when they reach position 48 (fully withdrawn). The licensee has implemented the improved GE overhaul procedure and test to assure proper installation of the inner filter. This has resulted in a significant reduction in events of uncoupled control rods.

Based on the foregoing, the staff finds the licensee's proposal to improve the wording of Technical Specification 4.3.B.1.b to be acceptable.

3.0 ENVIRONMENTAL QUALIFICATION

The staff has 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, the staff further concludes 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.

4.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) 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.

5.0 ACKNOWLEDGEMENT

The following staff members contributed to this evaluation:

T. M. Tongue
K. R. Ridgway
T. N. Tambling

Dated: November 3, 1983