Commonwealth Edison
One First National Plaza, Chicago, Illinois
Address Reply to: Post Office Box 767
Chicago, Illinois 60690

Regulatory File Cy.

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Mr. J. F. O'Leary, Director
Directorate of Licensing
Office of Regulation
U.S. Atomic Energy Commission
Washington, D.C. 20545

Subject: Proposed Change to Appendix A of DPR-19,
Dresden Unit 2 - AEC Dkt 50-237

Dear Mr. O'Leary:

Pursuant to Section 50.59 of 10 CFR Part 50 and Paragraph 3.B of Facility License DPR-19, Commonwealth Edison Company hereby submits a proposed change to Appendix A of DPR-19 (Dresden Unit 2). The purpose of this change is to modify the current Technical Specification concerning maximum allowable in-sequence control rod worth and control rod scram insertion time requirements.

The page changes to the Technical Specifications are attached and safety evaluation for the proposed change is given below.

In-Sequence Rod Worth

The evaluation of this change was presented in a similar proposed change for Dresden Unit 3, AEC Dkt 50-249, submitted by letter dated September 14, 1973.

Control Rod Scram Insertion Time

The evaluation of this proposed change was discussed in Dresden Special Report No. 29, dated July 2, 1973. This change involves more restrictive allowable scram insertion times; therefore no unreviewed safety concerns are created.

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Three signed originals and 37 copies of this proposed change are submitted for your use.

Very truly yours,

Byron Lee Jr

Byron Lee, Jr.
Vice-President

SUBSCRIBED and SWORN to
before me this 18th day
of October, 1973.

Brenda Pammer
Notary Public

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3.3 LIMITING CONDITION FOR OPERATION	4.3 SURVEILLANCE REQUIREMENT

C. Scram Insertion Times

1. The average scram insertion time, based on the de-energization of the scram pilot valve solenoids as time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.900
50	2.00
90	5.00

The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.398
20	0.954
50	2.120
90	5.300

2. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

C. Scram Insertion Times

1. After each refueling outage and prior to power operation with reactor pressure above 800 psig, all control rods shall be subject to scram-time tests from the fully withdrawn position. The scram times shall be measured without reliance on the control rod drive pumps.

2. At 16 week intervals, 50% of the control rod drives shall be tested as in 4.3.C.1 so that every 32 weeks all of the control rods shall have been tested. Whenever 50% of the control rod drives have been scram tested, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

3. 25 of the operable control rods, selected to be uniformly distributed throughout the core, shall be scram-time tested at full reactor pressure at the time intervals listed below following any outage exceeding 72 hours in duration: 1 week, 2 weeks, 4 weeks, 8 weeks, 16 weeks and continuing at 16 week intervals:
   a) If the mean 90% insertion time of the tested control rod drives increases by more than 0.25 seconds or if the mean insertion time exceeds 3.5 seconds, then an additional sample of 25 control rods, selected to be uniformly distributed throughout the core, shall be scram tested. If the mean 90% insertion time of the 50 selected control rod drives exceeds 4.25 seconds, then all operable drives will be tested. Subsequent testing shall revert to the original 25 control rods at the 1 week, 2 week, etc., sequence interval; and

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3.3 LIMITING CONDITION FOR OPERATION	4.3 SURVEILLANCE REQUIREMENT
3. a. Control rod withdrawal sequences shall be established so that maximum reactivity that could be added by dropout of any increment of any one control blade would not make the core more than 0.013 ΔK supercritical. b. Whenever the reactor is in the startup/hot standby or run mode below 10% rated thermal power, the rod worth minimizer shall be operable or a second licensed operator or other qualified technical station employee shall verify that the operator at the reactor console is following the control rod program.	3. The correctness of the control rod withdrawal sequence input to the RWM computer shall be verified after loading the sequence. Prior to the start of control rod withdrawal towards criticality, the capability of the Rod Worth Minimizer to properly fulfill its function shall be verified by the following checks: a. The RWM computer on line diagnostic test shall be successfully performed. b. Proper annunciation of the selection error of one out-of-sequence control rod shall be verified. c. The rod block function of the RWM shall be verified by withdrawing the first rod as an out-of-sequence control rod no more than to the block point.
4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second.	4. Prior to control rod withdrawal for startup or during refueling verify that at least two source range channels have an observed count rate of at least three counts per second.
5. During operation with limiting control rod patterns, as determined by the nuclear engineer, either: a. Both RBM channels shall be operable; or b. Control rod withdrawal shall be blocked; or c. The operating power level shall be limited so that the MCHFR will remain above 1.0 assuming a single error that results in complete withdrawal of any single operable control rod.	5. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod (s) and daily thereafter.

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indicative of a generic control rod drive problem and the reactor will be shutdown.

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 indicate an uncoupled condition.

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 in-sequence individual control rod or control rod segments which are withdrawn could not be worth enough to cause the core to be more than 0.013ΔK supercritical if they were to drop out of the core in the manner defined for the Rod Drop Accident. 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 supervised by the RWM or a second qualified station employe. This 0.013ΔK 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.

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Bases (cont'd)

These techniques are described in a topical report(1) and two supplements.(2)(3).

By using the analytical models described in those reports coupled with conservative or worst-case input parameters, it has been determined that for power levels less than 10% of rated power, the specified limit on in-sequence control rod or control rod segment worths will limit the peak fuel enthalpy to less than 280 cal/gm. Above 10% power even single operator errors cannot result in out-of-sequence control rod worths which are sufficient to reach a peak fuel enthalpy of 280 cal/gm should a postulated control rod drop accident occur.

(1) Paone, C.J., Stirn, R.C. and Wooley, J.A., "Rod Drop Accident Analysis for Large Boiling Water Reactors", NEDO-10527, March 1972.

(2) Stirn, R.C., Paone, C.J., and Young, R.M., "Rod Drop Accident Analysis for Large BWR's", Supplement 1 - NEDO-10527, July 1972.

(3) Stirn, R.C., Paone, C.J., and Haun, J.M., "Rod Drop Accident Analysis for Large BWR's Addendum No. 2, Exposed Cores", Supplement 2-NEDO 10527, January 1973.

The following conservative or worst-case bounding assumptions have been made in the analysis used to determine the specified 0.013 K limit on in-sequence control rod or control rod segment worths. Details of this analysis are contained in Reference 4.

a. A maximum inter-assembly local power peaking factor not expected to be reached during future reloads.

b. An end-of-cycle delayed neutron fraction.

c. A beginning-of-life Doppler reactivity feedback.

d. The technical specification rod scram insertion rate.

e. The maximum possible rod drop velocity (3.11 ft./sec.)

f. The design accident and scram reactivity shape function.

g. The minimum moderator temperature to reach criticality.

(4) Exhibit A attached to September 14, 1973 letter from Byron Lee, Commonwealth Edison Company, To J. F. O'Leary U.S. Atomic Energy Commission.

Bases (cont'd)

In most cases the worth in in-sequence rods or rod segments will be substantially less than 0.013ΔK. Further, the addition of 0.013ΔK worth of reactivity as a result of a rod drop and in a conjunction with the actual values of the other important accident analysis parameters described above would most likely result in a peak fuel enthalpy substantially less than the 280 cal/gm design limit. However, the 0.013ΔK limit is applied in order to allow room for future reload changes and ease of verification without repetitive Technical Specification changes.

Should a control drop accident result in a peak fuel energy content of 280 cal/gm less than 660 (7 x 7) fuel rods are conservatively estimated to perforate. This would result in an offsite dose well below the guideline value of 10CFR100. For 8 x 8 fuel, less than 850 rods are conservatively estimated to perforate with nearly the same consequences as for the 7 x 7 fuel case because of the rod power differences.

The Rod Worth Minimizer provides automatic supervision to assure that out of sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviations from planned withdrawal sequences. Ref. Section 7.9 SAR. It serves as a backup to procedural control of control rod worth. In the event that the Rod Worth

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