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Docket No. 50-249

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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 April 23, 1976, and August 23, 1976, and a supplement to the August 23, 1976 request dated October 19, 1976, the Commission has issued the enclosed Amendment No. 23 to Facility Operating License No. DPR-25 for Unit No. 3 of the Dresden Nuclear Power Station.

This amendment (1) authorizes operation with additional 8 x 8 fuel assemblies as replacement for existing fuel assemblies, (2) incorporates revised MCPR limits in response to the plant specific analysis for reload and (3) modifies License Condition 3.E to reflect End-of-Cycle scram reactivity conditions for reload 4 in response to your request dated August 23, 1976, and supplement dated October 19, 1976. The amendment also revises the remedial actions specified in the limiting conditions for operation related to MAPLHGR, LHGR and MCPR in response to your request dated April 23, 1976.

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

Sincerely,

Original signed by
Dennis L. Ziemann

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

Enclosures:
1. Amendment No. 23 to License No. DPR-25
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

Concurrence contingent on the SER being the same as that transmitted 11/2/76

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10/19/76:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-249

DRESDEN UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 23
License No. DPR-25

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

   A. The applications for amendment by Commonwealth Edison Company (the licensee) dated April 23, 1976, and August 23, 1976, and supplement dated October 19, 1976, comply 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.E of Facility License No. DPR-25 is hereby amended to read as follows:

   3.E Restrictions

   Reactor power level shall be limited to maintain pressure margin to the safety valve set points during the worst case pressurization transient. The magnitude of the power limitation, if any, and the point in the cycle at which it shall be applied is specified in the Reload No. 4 licensing submittal for Dresden Unit No. 3 (NEDO-21338). Plant operation shall be limited to the operating plant described therein.

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 4, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 23

FACILITY 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 noted. The changed areas on the revised pages are reflected by a marginal line.

Remove Pages	Insert Pages
ii of Table of Contents	ii of Table of Contents
21	21
81B	81B
	81B-1 - new
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Bases:

2.2 In compliance with Section III of the ASME Code, the safety valves must be set to open at no higher than 103% of design pressure, and they must limit the reactor pressure to no more than 110% of design pressure. Both the high pressure scram and safety valve actuation are required to prevent overpressurizing the reactor pressure vessel and thus exceeding the pressure safety limit. The pressure scram is actually a backup protection to the high flux scram which was analyzed in Section 4.4.3 of the SAR, re-examined in the Dresden 3 Second Reload License Submittal, September 14, 1973, and reanalyzed in "Dresden Station Special Report No. 29 Supplement B". If the high flux scram were to fail during a maximum pressure transient (also assuming failure of the turbine stop valve closure scram, failure of the bypass system to actuate and failure of the relief valves to open) the pressure would rise rapidly due to void reduction in the core. A high pressure scram would occur at 1060 psig. The pressure at the bottom of the vessel is about 1232 psig when the first safety valve opens and about 1272 psig when the last valve opens. Both values are clearly within the code requirements. Vessel dome pressure reaches less than 1298 psig with the peak at the bottom of the vessel less than 1327 psig. Therefore, the pressure scram and safety valve actuation provide adequate margin below the peak allowable vessel pressure of 1375 psig.

3.5 LIMITING CONDITION FOR OPERATION

I. Average Planar LHGR

During steady state power operation, the average linear heat generation rate (LHGR) of all the rods in any fuel assembly, as a function of average planar exposure, at any axial location, shall not exceed the maximum average planar LHGR shown in Figure 3.5.1. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.5 SURVEILLANCE REQUIREMENT

I. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$ 25% rated thermal power.

3.5 LIMITING CONDITION FOR OPERATION

J. Local LHGR

During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly, at any axial location, shall not exceed the maximum allowable LHGR as calculated by the following equation:

$$LHGR_{max} \leq LHGR_{d} \left[1-\left(\frac{\Delta P}{P}\right)_{max}\left(\frac{L}{LT}\right)\right]$$

$LHGR_{d}$ = Design LHGR = 17.5 Kw/ft. 7x7 fuel

= 13.4, 8x8 fuel

$\left(\Delta P/P\right)_{max}$ = Maximum power spiking penalty = 0.036 for 7x7 fuel and 0.026 for 8x8 fuel

LT = Total core length = 12 ft.

L = Axial position above bottom of core

If at any time during operation, it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

4.5 SURVEILLANCE REQUIREMENT

J. Linear Heat Generation Rate (LHGR)

The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% rated thermal power.

3.5 LIMITING CONDITION FOR OPERATION

K. Minimum Critical Power Ratio (MCPR)

During steady state operation, MCPR shall be greater than or equal to -

Unit 3

1.39 (7 x 7 fuel)

1.33 (8 x 8 fuel)

at rated power and flow. For core flows other than rated, these nominal values of MCPR shall be increased by a factor of K_f, where K_f is as shown in Figure 3.5-2.

If at any time during steady state power operation, it is determined that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits. For core flows other than rated, the MCPR shall be 1.32 times K_f where K_f is as shown in Figure 3.5-2.

4.5 SURVEILLANCE REQUIREMENTS

K. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during a reactor power operation at $\geq$ 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.5.

heat generation rate even if fuel pellet densification is postulated. The power spike penalty specified is based on that presented in Ref. (2) and assumes a linearly increasing variation in axial gaps between core bottom and top, and assumes with 95% confidence, that no more than one fuel rod exceeds the design LHGR due to power spiking. An irradiation growth factor of 0.25% was used as the basis for determining ΔP/P in accordance with Refs. (3) and (4).

K. Minimum Critical Power Ratio (MCPR)

The steady state values for MCPR specified in this Specification were selected to provide margin to accommodate transients and uncertainties in monitoring the core operating state as well as uncertainties in the critical power correlation itself. These values also assure that operation will be such that the initial condition assumed for the LOCA analysis, a MCPR of 1.18, is satisfied. For any of the special set of transients or disturbance caused by single operator error or single equipment malfunction, it is required that design analyses initialized at this steady state operating limit yield a MCPR of not less than that specified in Specification 1.1.A at any time during the transient assuming instrument trip settings given in Specification 2.1. For analysis of the thermal consequences of these transients, the limiting value of MCPR stated in this specification is conservatively assumed to exist prior to the initiation of the transients. The results apply with increased conservatism while operating with MCPR's greater than specified.

The most limiting transients with respect to MCPR are generally:

a) Rod withdrawal error

b) Turbine trip without bypass

c) Loss of feedwater heater

Several factors influence which of these transients results in the largest reduction in critical power ratio such as the specific fuel loading, exposure, and fuel type. The current cycles reload licensing submittal specifies the limiting transient for a given exposure increment for each fuel type. The values specified as the Limiting Condition for Operation are conservatively chosen as the most restrictive over the entire cycle for each fuel type. For Cycle 5, the operating limit has been increased by 0.07 for 8 x 8 fuel over the limit based on transient analyses to assure that boiling transition would not occur in a misloaded fuel bundle during steady state operation.

For core flow rates less than rated, the steady state MCPR is increased by the formula given in the Specification. This assure that the MCPR will be maintained greater than that specified in Specification 1.1.A even in the event that the motor-generator set speed controller causes the scoop tube positioner for the fluid coupler to move to the maximum speed position.

(2) Fuel Densification Effects on General on General Electric Boiling Water Reactor Fuel," Section 3.2.1, Supplement 6, Aug. 1973.

(3) USAEC Report, "Supplement 1 to the Technical Report on Densification of General Electric Reactor Fuels," Dec. 14, 1973.

(4) GE Planning and Development Memorandum #45, "Length Growth of BWR Fuel Elements", R. A. Proebsthe, October 1, 1973 (Proprietary).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 23 TO FACILITY OPERATING LICENSE NO. DPR-25

COMMONWEALTH EDISON COMPANY

DRESDEN UNIT NO. 3

DOCKET NO. 50-249

INTRODUCTION

By letter dated August 23, 1976 and supplement dated October 19, 1976, Commonwealth Edison Company (CECo) proposed to reload Dresden Unit No. 3 for Reload 4 (Cycle 5) and requested an amendment to Facility Operating License DPR-25. The amendment would modify the license and Technical Specifications to allow operation of the facility with:

(1) Additional 8 x 8 fuel assemblies as replacements for existing fuel assemblies.

(2) Revised MCPR limits based on plant specific analyses for Reload 4.

(3) License Condition 3.E (power restrictions) modified to reflect the actual scram reactivity condition predicted for Cycle 5.

By letter dated April 23, 1976, in response to our request dated March 1, 1976, Commonwealth Edison requested an amendment which would incorporate more specific limiting conditions of operation related to MAPLHGR, LHGR and MCPR.

During our review of the proposed technical specifications we determined that certain changes were necessary to conform with Regulatory requirements. These changes have been accepted by CECo.

DISCUSSION

The reference core loading for Dresden 3, Reload 4, consists of 340 initial 7 x 7 fuel assemblies, 52 Reload 1 7 x 7 fuel assemblies, 44 Reload 2 8 x 8 fuel assemblies, 140 Reload 3 8 x 8 fuel assemblies and 148 Reload 4 8 x 8 fuel assemblies. The use of 8 x 8 fuel assemblies for reloads was previously evaluated and approved by Change No. 16 dated March 25, 1974 and Amendment No. 5 dated August 29, 1975.

The Reload 4 fuel uses the same enrichments as were used in previous 8 x 8 fuel reloads. The mechanical design is also identical to the design approved for previous reloads with the exception that the Reload 4 8 x 8 fuel uses an improved water rod design.[4]

Our safety evaluation for this reload (Reload No. 4) for Dresden 3 is based on the licensee's application as amended[1], [2], [3] and on information contained in a GE topical report NEDO-20360[4] referenced in the application.

EVALUATION

Nuclear Characteristics

The information presented in the licensing submittal for the reconstituted core closely follows the guidelines of Appendix A of Reference 4. Although later supplements to this report are undergoing review by the staff, this topical has been found acceptable for use for reactors containing 8 x 8 reload fuel. The design reference core reload is based on the use of 60 Reload 4 8 x 8 bundles having a bundle average enrichment of 2.50 weight percent U-235 (8D250 bundles) and 88 Reload 4 bundles having a bundle average enrichment of 2.62 weight percent U-235. The core contains a total of 724 fuel bundles. Thus, about 20 percent of the fuel bundles are being replaced for this reload. The loading

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1. Commonwealth Edison letter (Bolger) to NRC (Rusche), "Dresden Station Unit 3 Proposed Amendment to Facility Operating License No. DPR-25, NRC Docket No. 50-249", dated August 23, 1976.

2. Commonwealth Edison letter (Abrell) to NRC (Ziemann), "Dresden Station Unit 3 Supplement A to Reload 4 License Amendment to Facility Operating License No. DPR-25, NRC Docket No. 50-249", dated October 19, 1976.

3. "GE/BWR Reload No. 4 Licensing Submittal for Dresden Nuclear Power Station Unit 3", NEDO-21338 Class I, July 1976.

4. "GE/BWR Generic Reload Licensing Application for 8 x 8 fuel", NEDO-20360 Supplement 4 to Revision 1, dated April 1976.

pattern for Reload 4 may be described as follows: (1) the two rows and columns of fuel bundles which intersect at the center of the core will not contain any Reload 4 fuel, (2) the higher enrichment reload bundles are loaded in the interior of the core while the lower enrichment reload bundles are loaded near the periphery of the core, (3) for all the core (except as noted in 1 above) one fuel bundle in a four bundle array surrounding a control rod will be replaced. The data in Reference 3 indicate that the nuclear characteristics of the Reload 4 8 x 8 fuel bundles are similar to those previously loaded. Thus, the total control system reactivity worth, temperature, and void dependent behavior of the reconstituted core will not differ significantly from those values which were previously reported for Dresden Unit No. 3.

The shutdown margin of the reconstituted core meets the Technical Specification requirement that the core be at least 0.0025Δk subcritical in the most reactive operating state with the control rod of greatest reactivity worth fully withdrawn and with all other control rods fully inserted. A minimum shutdown margin during Cycle 5 is 0.0215Δk and occurs at a core average exposure of 11,260 Mwd/t.

The information presented in the application[3/] indicates that a boron concentration of 600 ppm in the moderator will make the reactor subcritical by at least 0.03Δk at 20°C, xenon free. Therefore, the alternate shutdown requirement of the General Design Criteria is met by the Standby Liquid Control System.

The Technical Specification requirement for the storage of fuel for Dresden Unit No. 3 is that the effective multiplication factor, k_{eff}, of the fuel as stored in the fuel storage rack is equal to or less than 0.90 for normal conditions. This is achieved if the uncontrolled k_{eff} of a single fuel bundle is less than 1.30[4/] at 65°C. The 8 x 8 8D250 and 8D262 fuel bundles, at both the zero exposure and the peak reactivity point, have a k_{eff} less than 1.26 and, therefore, the fuel storage requirement for Dresden Unit No. 3 is acceptable.

The scram reactivity curves used in the transient analysis for Cycle 5 are shown in Figure 6-6 of Reference 2. The GE generic "B" curves and end of cycle curve used in previous transient analyses have been replaced by plant specific curves. However, the changes are slight. Also, these scram curves are multiplied by a design conservatism factor of 0.8 for use in the transient analyses.

Based on our review of the information presented in the Dresden Unit No. 3 submittals and the generic 8 x 8 reload report (Reference 4), we conclude that the nuclear characteristics (e.g. scram reactivity, void coefficient of reactivity and Doppler coefficient of reactivity) and performance of the reconstituted core for the Reload 4 fuel Cycle 5 will not differ significantly from previous Dresden Unit No. 3 fuel cycles and is acceptable.

Mechanical Design

The Reload 4 fuel has the same mechanical configuration and fuel bundle enrichment as the 8D250 and 8D262 fuel assemblies described in the 8 x 8 generic reload report (Reference 4). This generic report, which is under review has been found acceptable for use for reactors containing 8 x 8 reload fuel. Reload 4 incorporates the improved water rod design described in subsection 3.1 of Reference 4. The staff has reviewed the water rod design on a generic basis[5] and determined that the design is acceptable.

Finger springs are not used in the Reload 4 fuel assemblies. The criteria used to determine the necessity for finger springs is bypass flow. Design bypass flow should be maintained between 10% and 12% of the total core flow. The bypass flow for Dresden 3 with no finger springs on Reload 4 fuel is 11.7 percent.[2] On the basis of our review of the generic 8 x 8 reload design in similar plants, we conclude that the mechanical design of Dresden Unit No. 3 Reload 4 fuel is acceptable.

Thermal-Hydraulics

The GE generic 8 x 8 fuel reload topical report[4] and GETAB[6] are referenced to provide the description of the thermal-hydraulic methods which were used to calculate the thermal margins. Application of GETAB involves:

(1) establishing the fuel damage safety limit,

(2) establishing limiting conditions of operation such that the safety limit is not exceeded for normal operation and anticipated transients, and

(3) establishing limiting conditions for operation such that the initial conditions assumed in the accident analyses are satisfied.

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5. Technical Report on the General Electric Company 8 x 8 Fuel Assembly, dated February 5, 1974, by the Directorate of Licensing.

6. "GE/BWR Thermal Basis (GETAB): Data, Correlation and Design Application", NEDO-10958, 74NED9, Class I, November 1973.

We have evaluated and report in the following section the Dresden Unit No. 3 Reload 4 thermal margins based on the GETAB report [6] and plant specific input information provided by the licensee.

Fuel Cladding Integrity Safety Limit Minimum Critical Power Ratio (MCPR)

A critical power ratio (CPR) is defined as the ratio of that assembly power which causes some point in the assembly to experience transition boiling to the assembly power at the reactor condition of interest. The minimum critical power ratio (MCPR) is the critical power ratio corresponding to the most limiting fuel assembly in the core.

The fuel cladding integrity safety limit MCPR is 1.06. It is based on the GETAB statistical analysis which assures that 99.9% of the fuel rods in the core are not expected to experience boiling transition during abnormal operational transients. The uncertainties in the core and system operating parameters and the GEXL correlation, combined with the relative bundle power distribution in the core form the basis for the GETAB statistical determination of the safety limit MCPR. The tabulated list of uncertainties for Dresden Unit No. 3 Cycle 5 are the same or more conservative than those used in GETAB (Table 4-2 Reference 4). For example, the standard deviation for the TIP readings uncertainty for the subject reload is 8.7% whereas the GETAB NEDO-10958 report shows 6.3%. The increase in uncertainty for the subject reload is a consequence of the increased uncertainty in the measurement of power in a reload core. In both cases the TIP reading uncertainties are based on a symmetrical planar power distribution.

The generic core selected for the GETAB statistical analysis is a typical 251/764 core while the Dresden Unit No. 3 is a 251/724 core. The generic GETAB statistical analysis results are conservative since the bundle power distribution used for the GETAB application has more high power bundles than the distribution expected during the fifth cycle of operation of the Dresden Unit No. 3 reactor. This results in a conservative value of the MCPR which meets the 99.9% criterion. We conclude that the proposed fuel integrity safety limit, a MCPR of 1.06 is acceptable for Dresden Unit No. 3 Fuel Cycle 5.

Operating Limit MCPR

Various transient events will reduce the MCPR below the operating MCPR. To assure that the fuel cladding integrity safety limit (MCPR of 1.06) is not exceeded during anticipated abnormal operational transients, the most

limiting transients have been analyzed to determine which one results in the largest reduction in critical power ratio. The licensee, in Reference 3 has stated that "all transients which are the basis of the existing license were reviewed and those transients which have been limiting in the past with respect to the safety margins and are significantly sensitive to the core transient parameter deviations were reanalyzed." The licensee has submitted the results of analyses of those transients which produce a significant decrease in MCPR (Reference 3, Table 4-1). Among the transients evaluated were turbine trip without bypass, feedwater temperature decrease and rod withdrawal error. The most limiting transient from rated conditions for the 7 x 7 fuel is a Rod Withdrawal Error (RWE) which has a maximum reduction in MCPR of 0.33. The most severe transient for the 8 x 8 fuel is a turbine trip without bypass at end of cycle (EOC), which has a maximum Δ MCPR of 0.20. Addition of these reductions in CPR to the safety limit MCPR (1.06) gives the minimum operating limit MCPR for each fuel type required to avoid violation of the safety limit, should this limiting transient occur. Therefore, the operating limit MCPR's are 1.39 for 7 x 7 fuel and 1.26 for 8 x 8 fuel. However, these limits would be further increased to 1.33 for 8 x 8 fuel to compensate for the reduction in CPR that would result from a fuel loading error as discussed later.

The transient analyses were evaluated with scram reactivity insertion rates that included a design conservatism factor of 0.80. The design conservatism factor for the void coefficient used was 1.33 and the design conservatism factor for Doppler coefficient was 0.90. The initial conditions [3] used for the worst operational transient analyses were equal to or greater than the established operating limit MCPR for each fuel type.

A GE study[6] has shown that the required operating MCPR varies with the axial and local power peaking distribution. Axial peaking in the middle or upper portion of the core results in higher required MCPR's than peaking in the lower portion of the core. In the analyses the axial power peak was assumed to be representative of beginning-of-cycle conditions and to be located in the upper portion of the core.

The R-factors, which are a function of the local power peaking, assumed in the analyses are also representative of beginning-of-cycle conditions. The values assumed are 1.100 for 7 x 7 fuel and 1.094 for 8 x 8 fuel. During the cycle the local peaking and therefore the R-factor is reduced while the peak in the axial shape moves toward the bottom of the core. Although the operating limit MCPR would be increased by approximately 1% by the reduced end-of-cycle R-factor, this is offset by the reduction in MCPR resulting from the relocation of the axial peak to below the midplane.

Conservatism was applied in the determination of the required operating limit MCPR because the assumed axial and local peaking were representative of the beginning of the fuel cycle. This is the worst consistent set of axial and local peaking.

Analyses have shown that the operating limit MCPR's of 1.39 for 7 x 7 fuel and 1.26 for 8 x 8 fuel assure that the fuel cladding integrity safety limit is not exceeded during anticipated abnormal operational transients. Hence, we conclude that the operating limit MCPR's of 1.39 for 7 x 7 fuel and 1.26 for 8 x 8 fuel based on transient analyses are acceptable.

Turbine Trip Without Bypass

The turbine trip without bypass transient produces an isolation of the reactor coolant system which results in the most severe pressure transient. The pressure increase is due to closure of the turbine stop valves and obstruction of steam flow. The pressure increase also causes a significant void reduction which causes an increase in reactivity and a rapid increase in neutron flux. The rapid increase in neutron flux also causes a decrease in the critical power ratio. The results of the analysis are affected by the scram reactivity curve (decrease in reactivity as a function of control rod insertion). The scram reactivity curve becomes less favorable with core burnup. The analyses show that a pressure transient initiated from 100% power is acceptable until 2000 MwD/tonne before the end of the cycle. After that, because of a less favorable scram reactivity curve, the reactor power has to be limited to 98% to maintain an acceptable pressure margin during the transient. End-of-cycle power deratings as a result of changes of scram reactivity curves during the cycles has also occurred in previous operating cycles at Dresden No. 3. The reduction in CPR resulting from a turbine trip without bypass in Cycle 5 is 0.18 for 7 x 7 fuel and 0.20 for 8 x 8 fuel. Turbine trip is the limiting transient for 8 x 8 fuel. Although the largest reduction in CPR for 8 x 8 fuel occurs at EOC, the most restrictive value is chosen to apply as the Limiting Condition for Operation (LCO) over the entire cycle.

Rod Withdrawal Error

The control rod withdrawal error transient is the limiting transient for the 7 x 7 fuel during Cycle 5 with the Rod Block Monitor (RBM) subsystem setting at 107% of its initial level. The rod withdrawal error transient is discussed in Reference 3 in terms of worst case conditions. Assumptions and descriptions of the rod withdrawal event are given in Reference 4. The information in these two references indicates that the local power range monitor subsystem (LPRM's) will detect high local powers and alarm. However, if the operator ignores the LPRM alarm,

the rod block monitor subsystem (RBM) will stop the rod withdrawal while the critical power ratio is still equal to or greater than the 1.06 MCPR safety limit and the cladding is under the one percent plastic strain limit. We conclude that the analysis performed for this localized transient and the consequences of this localized transient are acceptable.

Loss of a Feedwater Heater

The loss of a feedwater heater is analyzed because it constitutes the most limiting cool water injection transient. A loss of 100°F of feedwater heating is assumed. This transient results in a reduction in CPR of 0.15 for both 7 x 7 and 8 x 8 fuels. This transient is not limiting for either type fuel.

Operating MCPR Limits for Less Than Rated Power and Flow

For operation at less than rated power and flow, the limiting transient is recirculation pump speed control failure. To assure that the safety limit MCPR is not violated by this transient, the licensee will conform to Technical Specification limiting conditions for operation. This requires that minimum MCPR values for less than rated flow are the rated flow values (1.39 for 7 x 7 fuel and 1.26 for 8 x 8 fuel) multiplied by the respective k_f factors appearing in Figure 3.2-2 of the Technical Specifications. The k_f factor curves were generically derived and assure that the most limiting transient occurring at less than rated flow will not exceed the safety limit MCPR of 1.06. We conclude that the calculated consequences of the anticipated abnormal transients do not violate the thermal and plastic strain limits of the fuel or the pressure limits of the reactor coolant boundary.

Accident Analysis

Overpressure Analysis

Reference 3 presents the results of an overpressure analysis to demonstrate that an adequate margin exists below the ASME code allowable vessel pressure of 110% of the vessel design pressure. The transient analyzed was closure of all main steam isolation valves with high neutron flux scram. The analysis was performed for 100% power with the end-of-cycle scram reactivity insertion rate curve (which is more conservative than the beginning-of-cycle curve), void reactivity applicable for this reload, no credit for the relief function of the safety relief valves, with all safety valves operative. The results of this analysis indicate that the

peak pressure at the vessel bottom would be 1294 psig. Although a specific analysis was not performed for Dresden Unit No. 3 with one safety valve failed, it is stated in Reference 2 that the sensitivity study of Reference 7 if applied to Dresden Unit No. 3 would show a smaller pressure increase for one failed safety valve than is the case for the typical high power density BWR for which the sensitivity study was performed. If the valve of 20 psig increases for one safety valve failed on a high power density BWR is applied to Dresden Unit No. 3, the maximum pressure for the overpressure transient is 1314 psig. This results in about a 61 psig margin below the code allowable limit of 1375 psig, which is acceptable to the staff.

Fuel Loading Error

Fuel loading errors are discussed in Reference 4 for an 8 x 8 fuel bundle placed in an improper location or rotated 180 degrees in a location near the center of the core. The information in Reference 3 indicates that a fuel loading error results in a peak linear heat generation (LHGR) of 16.3 kW/ft and a minimum critical power ratio (MCPR) of 0.99 in the misplaced 8 x 8 (2.62% enrichment) fuel bundle during steady state operation. Since the MCPR for the bundle is lower than the core wide safety limit of 1.06, it is expected that some of the fuel rods in the bundle would experience boiling transition and must therefore be presumed to fail. Fuel bundles adjacent to a misplaced fuel bundle will not be significantly affected by the presence of a misplaced bundle.

The fuel loading accident is being generically reviewed by the NRC staff and generic resolution is anticipated. In the interim the licensee has agreed to increase the operating limit MCPR to 1.33 for 8 x 8 fuel assemblies, respectively, thereby assuring that the safety limit of 1.06 will not be violated by this accident. This will keep the rod bundle from boiling transition during steady state operation even if the worst misloading error should occur. We have concluded that this is acceptable.

Control Rod Drop Accident

The control rod drop accident for the Dresden Unit No. 3 re-loaded core is within the bounding analysis presented in Reference 4. The Doppler

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7. Letter to I. F. Stuart (GE) to V. Stello (NRC), "Code Overpressurization Protection Analysis-Sensitivity of Peak Vessel Pressure G Valve 0 Penalty", dated December 23, 1975.

coefficient of reactivity, the accident reactivity shape and magnitude function, and the rod drop scram reactivity functions are compared with the technical bases presented in Reference 4. This analysis is performed for Doppler coefficients of reactivity at the beginning of the Reload 4 fuel cycle, zero void fraction, and at both cold (20°C) and hot (286°C) startup conditions. Incremental bank withdrawal is also assumed. It is shown by Figures 6-1, 6-2, 6-3, 6-4 and 6-5 of Reference 3 that the maximum values of the parameters for this reloaded core will not exceed the bounding values. Therefore, we conclude that the consequences of a control rod drop accident from any insequence control rod during startup will be below the design limit of 280 cal/gm.

Main Steam Line Break, Refueling Accident

The analyses of the Main Steam Line Break and the Refueling Accident are covered by the generic analyses given in Reference 4. Based on previous review of the referenced material the staff concludes that the results provided by the generic analyses are applicable to Dresden Unit No. 3 Cycle 5 and are acceptable.

Loss-of-Coolant Accident

The analyses of Loss-of-Coolant accident for the Dresden No. 3 Cycle 4 fuel were performed by the licensee using the analytical models and associated assumptions given in Reference 4 which were approved by the NRC. The results of these analyses are in conformance with the requirements of 10 CFR 50.46. The reload fuel for Cycle 5 is the same type as previously licensed for operation in Cycle 4. Based upon our review, we conclude that the operation of the reactor will meet the requirement of 10 CFR 50.46 with operation limited to the maximum average planar linear heat generation rates (MAPLHGR) presently in the Dresden Unit No. 3 Technical Specifications. The operating minimum critical power ratio (MCPR) limit determined from analyses of anticipated abnormal operational transients is more limiting than that determined from the analysis of loss-of-coolant accidents.

We have concluded that the accident analyses for Reload 4 have been performed in accordance with methods acceptable to the NRC staff and demonstrate that the consequences of postulated accidents are acceptable.

Physics Startup Testing

As part of our review of Reload 4, the licensee was requested to provide a description of the Physics Startup testing program. In response to our request, this program was described in Reference 2. The list of startup tests performed by CECo, along with the test required to assure compliance with the Technical Specifications, provide an acceptable Startup Physics testing program.

Technical Specification Changes

The proposed Technical Specification changes to operating limit MCPR are based on a plant specific analysis of the limiting transients for Cycle 5 with the appropriate design conservatism factors. A MCPR of 1.39 for 7 x 7 fuel and 1.26 for 8 x 8 fuel based on this analysis is acceptable to the staff for Cycle 5. In the interim, prior to further analyses of the consequences of fuel loading errors, the MCPR limiting conditions for operation would be conservatively increased to 1.33 for 8 x 8 fuel.

The proposed amendment to require derating to 98% power at 2000 Mwd/t before the end of the cycle (2300 to 2400 after the beginning of the cycle) to maintain the margin to safety value setpoints based on the analysis of the turbine trip transient for 8 x 8 fuel is acceptable to the staff for Cycle 5.

By letter dated March 1, 1976, the NRC requested Commonwealth Edison to include, in the Technical Specifications for Average Planar Linear Heat Generation Rate (APLHGR), Linear Heat Generation Rate (LHGR), and Minimum Critical Power Ration (MCPR), explicit remedial actions to be taken if the specifications are exceeded. The revised specifications would require, upon exceeding a limit, the initiation of remedial action within 15 minutes to restore conditions to within the prescribed limit. If operation was not within the specified limits within two (2) hours, the proposed specifications would require that the reactor be placed in Cold Shutdown within 36 hours. Current Dresden Unit No. 3 Technical Specifications for APLHGR, LHGR, and MCPR do not specify time limits for remedial action.

The NRC staff has reviewed the proposed technical specifications regarding remedial action for APLHGR, LHGR, and MCPR limits. We conclude that the revised specifications are in accordance with the provisions of 10 CFR Part 50 §50.36(c)(2), which permits a limited period of time for operator action to be taken to restore plant parameters within operating limits rather than requiring that the plant be immediately shut down. In addition, the revision would improve the APLHGR, LHGR, and MCPR specifications by placing more specific requirements on the operator. On this basis, the proposed technical specifications are acceptable.

ENVIRONMENTAL CONSIDERATION

We have determined that the amendment does 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 amendment involves 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 this amendment.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendment 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, the amendment does 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: November 4, 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-249

COMMONWEALTH EDISON COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 23 to Facility Operating License No. DPR-25, issued to Commonwealth Edison Company (the licensee), which revised Technical Specifications for operation of the Dresden Nuclear Power Station Unit No. 3 (the facility) located in Grundy County, Illinois. The amendment is effective as of its date of issuance.

The amendment (1) authorized operation with additional 8 x 8 fuel assemblies as replacements for existing fuel assemblies, (2) incorporated revised MCPR limits in response to the plant specific analysis for reload 4, (3) modified License Condition 3.E to reflect end-of-cycle scram reactivity conditions for reload 4, and (4) revised the remedial actions specified in the Technical Specifications related to core thermal-hydraulic limits (MAPLHGR, LHGR, and MCPR).

The applications for the amendment comply 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 amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment 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 this amendment.

For further details with respect to this action, see (1) the application for amendment dated April 23, 1976, (2) the application for amendment dated August 23, 1976, and a supplement thereto dated October 19, 1976, (3) Amendment No. 23 to License No. DPR-25, and (4) 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 (3) and (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 fourth day of November, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann

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