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Docket No. 50-249

Commonwealth Edison Company
ATTN: Mr. C. Reed
Assistant Vice President
P. O. Box 767
Chicago, Illinois 60690

Gentlemen:

The Commission has issued the enclosed Amendment No. 34 to Facility Operating License No. DPR-25 for Dresden Nuclear Power Station Unit No. 3. The amendment consists of changes to the Technical Specifications in response to your application dated December 28, 1977, as supplemented March 27, 1978.

This amendment (1) authorizes operation with additional 8 x 8 fuel assemblies (2) incorporates revised MCPR limits in response to the plant specific analysis for reload 3 and (3) modifies License Condition 3.E to more concisely state the plant operating plan described in the licensee's reload submittal dated December 28, 1977.

In addition to the authorization provided by this amendment, the safety evaluation (enclosed) sets forth (page 7) your commitment to submit in the future either a Technical Specification change or a reanalysis, well before 30,000 MWD/t burnup, on the limits for 8D250 fuel linear heat generation rate.

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

Sincerely,

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 34 to DPR-25
2. Safety Evaluation
3. Notice

cc w/enclosures: see next page

*SEE PREVIOUS YELLOW FOR CONCURRENCES

OFFICE	ORB#3	ORB#3	OELD	ORB#3		
SURNAME	*SSheppard	*RBevan:acr	*DSwanson	GLear		
DATE	4/26/78	4/26/78	5/1/78	5/ 3 /78		

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Docket No. 50-249

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ATTN: Mr. C. Reed
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P. O. Box 767
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Gentlemen:

The Commission has issued the enclosed Amendment No. to Facility Operating License No. DPR-25 for Dresden Nuclear Power Station Unit No. 3. The amendment consists of changes to the Technical Specifications in response to your application dated December 28, 1977, as supplemented March 27, 1978.

This amendment (1) authorizes operation with additional 8 x 8 fuel assemblies (2) incorporates revised MCPR limits in response to the plant specific analysis for reload 3 and (3) modifies License Condition 3.E to more concisely state end-of-cycle scram reactivity conditions for core Reload No. 5.

In addition to the authorization provided by this amendment, the safety evaluation (enclosed) sets forth (page 7) your commitment to submit in the future either a Technical Specification change or a reanalysis, well before 30,000 MWD/t burnup, on the limits for 8D250 fuel linear heat generation rate.

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

Sincerely,

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. to DPR-25
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3. Notice

D A subject to correction of typos on p. 1 of amendment & p. 7 of SE.

cc w/enclosures:	See page 2	ORB #3	ORB #3	OELD	ORB #3
OFFICE ➤					
SURNAME ➤			SSheppard	RBevan:mjf	D Swanson	GLear
DATE ➤			4/26/78	4/26/78	5/1/78	/ /78

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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. 34
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 December 28, 1977, as supplemented on March 27, 1978, 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;

   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 paragraphs 3.B and 3.E of Facility License No. DPR-25 are hereby amended to read as follows:

   3.B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 34, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3.E. Restrictions

Reactor power level shall be limited to maintain pressure margin to the safety valve setpoints 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 licensee's reload submittal for Dresden Unit 3 (NEDO-24074, dated November 1977). Plant operation shall be limited to the operating plan described therein, with subsequent operation in the coastdown mode permitted to 40% power using full recirculation flow.

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

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 3, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 34

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

Remove Pages	Insert Pages
20	20
21	21
42	42
48	48
81D	81D
85B	85B
87	87
92	92

Bases:

1.2 The reactor coolant system integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this system be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1325 psig as measured by the vessel steam space pressure indicator is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The 1375 psig value is derived from the design pressures of the reactor pressure vessel, coolant system piping and isolation condenser. The respective design pressures are 1250 psig at 575°F, 1175 psig at 560°F, and 1250 psig at 575°F. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: ASME Boiler and Pressure Vessel Code, Section III for the pressure vessel and isolation condenser and USASI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10% over design pressure (110% X 1250 = 1375 psig), and the USASI Code permits pressure transients up to 20% over the design pressure (120% X 1175 = 1410 psig). The Safety Limit pressure of 1375 psig is referenced to the lowest elevation of the primary coolant system.

The design basis for the reactor pressure vessel makes evident the substantial margin of protection against failure at the safety pressure limit of 1375 psig. The vessel has been designed for a general membrane stress no greater than 26,700 psi at an internal pressure of 1250 psig; this is a factor of 1.5 below the yield strength of 40,100 psi at 575°F. At the pressure limit of 1375 psig, the general membrane stress will only be 29,400 psi, still safe below the yield strength.

The relationships of stress levels to yield strength are comparable for the isolation condenser and primary system piping and provide a similar margin of protection at the established safety pressure limit.

The normal operating pressure of the reactor coolant system is 1000 psig. For the turbine trip or loss of electrical load transients, the turbine trip scram or generator load rejection scram, together with the turbine bypass system, limit the pressure to approximately 1100 psig (4). In addition, pressure relief valves have been provided to reduce the probability of the open discharge safety valves operating in the event that the turbine bypass should fail. These valves and the neutron flux scram limit the reactor pressure to a value which is at least 25 psi below the setting of the first safety valve. Finally, the safety valves are sized to keep the reactor coolant system pressure below 1375 psig with no credit taken for the relief valves during the postulated full closure of all MSIV's without direct (valve position switch) scram. Credit is taken for the neutron flux scram, however. The pressure at the bottom of the vessel peaks at less than 1325 psig. The indirect flux scram and safety valve actuation, therefore, provide adequate margin below the peak allowable vessel pressure of 1375 psig.

Reactor pressure is continuously monitored in the control room during operation on a 1500 psi full scale pressure recorder.

---

(4) SAR, Section 11.2.2.

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 neutron flux 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 available as a backup protection to the high flux scram which was analyzed in References (5) and (6) and is reexamined in the Reload Licensing Submittal for each subsequent cycle. If the high flux scram were to fail, a high pressure scram would occur at 1060 psig.

(5) SAR, Section 4.4.3

(6) Special Report No. 29 and Supplement B thereto.

INSTRUMENTATION THAT INITIATES ROD BLOCK

Minimum No. of Operable Inst. Channels Per Trip System(1)	Instrument	Trip Level Setting
1	APRM upscale (flow bias) (7)	$\leq [.65W + 43] \left[\frac{LTPF}{TPF}\right]$ (2)
* 1	APRM upscale (refuel and Startup/Hot Standby mode)	$\leq$ 12/125 full scale
2	APRM downscale (7)	$\geq$ 3/125 full scale
1	Rod block monitor upscale (flow bias) (7)	$\leq [.65W + 42]$ (2)
1	Rod block monitor downscale (7)	$\geq$ 5/125 full scale
3	IRM downscale (3)	$\geq$ 5/125 full scale
3	IRM upscale	$\leq$ 108/125 full scale
* 3	IRM detector not fully inserted in the core	
2(5)	SRM detector not in startup position	(4)
2(5)(6)	SRM upscale	$\leq 10^5$ counts/sec

Two sensors on the isolation condenser supply and return lines are provided to detect the failure of isolation condenser line and actuate isolation action. The sensors on the supply and return sides are arranged in a 1 out of 2 logic and, to meet the single failure criteria, all sensors and instrumentation are required to be operable. The trip settings of 20 psig and 32" of water and valve closure time are such as to prevent uncovering the core or exceeding site limits. The sensors will actuate due to high flow in either direction.

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI piping. Tripping of this instrumentation results in actuation of HPCI isolation valves, i.e., Group 4 valves. Tripping logic for this function is the same as that for the isolation condenser and thus all sensors are required to be operable to meet the single failure criteria. The trip settings of 200°F and 300% of design flow and valve closure time are such that core uncovery is prevented and fission product release is within limits.

The instrumentation which initiates ECCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion the Specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR does not approach 1.06. The trip logic for this function is 1 out of n; e.g., any trip on one of the six APRM's, 8 IRM's, or 4 SRM's will result in a rod block. The minimum instrument channel requirements assure sufficient instrumentation to assure the single failure criteria are met. The minimum instrument channel requirements for the RBM may be reduced by one for a short period of time to allow for maintenance, testing, of calibration. This time period is only ~3% of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal.

The APRM rod block function is flow biased and prevents a significant reduction in MCPR expecially during operation at reduced flow. The ARPM provides gross core protection; i.e., limits the gross core control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than 1.06.

The APRM rod block function which is set at 12% of rated power is functional in the refuel and Startup/Hot Standby mode. This control rod block provides the same type of protection in the Refuel and Startup/Hot Standby mode as the APRM flow biased rod block does in the run mode; i,e., it prevents MCPR from decreasing below 1.06 during control rod withdrawals and prevents control rod withdrawal before a scram is reached.

The RBM rod block function provides local protection of the core, i.e., the prevention of transition boiling in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern. The trip point is flow biased. The worse case single control rod withdrawal error has been analyzed and the results show that with the specified trip settings rod withdrawal is blocked before the MCPR reached 1.06 thus allowing adequate margin.

Below 30 percent power, the worst case withdrawal of a single control rod results in a MCPR greater than 1.06 without rod block action. Thus, below this power level it is not required.

3.5 LIMITING CONDITION FOR OPERATION

Minimum Critical Power Ratio (MCPR)

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

Unit 3

1.39 (7 x 7 fuel)

1.39 (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.

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) Load rejection or 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 of Operation are conservatively chosen as the most restrictive over the entire cycle for each fuel type. For Cycle 6, the operating limit has been increased by 0.10 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).

3.6 LIMITING CONDITION FOR OPERATION	4.6 SURVEILLANCE REQUIREMENT

3.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the operating status of the reactor coolant system.

Objective:

To assure the integrity and safe operation of the reactor coolant system

Specification:

A. Thermal Limitations

1. Except as indicated in 3.6.A.2 below, the average rate of reactor coolant temperature change during normal heatup or cooldown shall not exceed 100°F/hr when averaged over a one-hour period.

2. A step reduction in reactor coolant temperature of 240°F is permissible so long as the limit in Specification 3.6.A.3 below is met.

3. At all times, the shell flange to shell temperature differential shall not exceed 140°F.

4. Core thermal power shall not exceed 25% of rated thermal power without forced recirculation.

4.6 PRIMARY SYSTEM BOUNDARY

Applicability:

Applies to the periodic examination and testing requirements for the reactor coolant system.

Objective:

To determine the condition of the reactor coolant system and the operation of the safety devices related to it.

Specification:

A. Thermal Limitations

1. During heatups and cooldowns the following temperatures shall be permanently recorded at 15 minute intervals:

   a. reactor vessel shell
   b. reactor vessel shell flange
   c. recirculation loops A and B

2. The temperatures listed in 4.6.A.1 shall be permanently recorded subsequent to a heatup or cooldown at 15 minute intervals until three consecutive readings are within 5 degrees of each other.

Bases:

A. Thermal Limitations – The reactor vessel design specification requires that the reactor vessel be designed for a maximum heatup and cooldown rate of the contained fluid (water) of 100°F per hour averaged over a period of one hour. This rate has been chosen based on past experience with operating power plants. The associated time periods for heatup and cooldown cycles when the 100°F per hour rate is limiting provides for efficient, but safe, plant operation.

The reactor vessel manufacturer has designed the vessel to the above temperature criterion. In the course of completing the design, the manufacturer performed detailed stress analysis. This analysis includes more severe thermal conditions than those which would be encountered during normal heating and cooling operations.

Specific analyses were made based on a heating and cooling rate of 100°F/hour applied continuously over a temperature range of 100°F to 550°F. Because of the slow temperature-time response of the massive flanges relative to the adjacent head and shell sections, calculated temperatures obtained were 500°F (shell) and 360°F (flange) (140°F differential). Both axial and radial thermal stresses were considered to act concurrently with full primary loadings. Calculated stresses were within ASME Boiler and Pressure Vessel Code Section III stress intensity and fatigue limits even at the flange area where maximum stress occurs.

The flange metal temperature differential of 140°F occurred as a result of sluggish temperature response and the fact that the heating rate continued over a 450°F coolant temperature range.

The uncontrolled cooldown rate of 240°F was based on the maximum expected transient over the lifetime of the reactor vessel. This maximum expected transient is the injection of cold water into the vessel by the high pressure coolant injection subsystem. This transient was considered in the design of the pressure vessel and five such cycles were considered in the design. Detailed stress analyses were conducted to assure that the injection of cold water into the vessel by the HPCI would not exceed ASME stress code limiations.

Specification 3.6.A.4 increases margin of safety for thermal-hydraulic stability and startup of recirculation pump.

B. Pressurization Temperature – The reactor coolant system is a primary barrier against the release of fission products to the environs. In order to provide assurance that this barrier is maintained at a high degree of integrity, restrictions have been placed on the operating conditions to which it can be subjected.

The nil-ductility transition (NDT) temperature is defined as the temperature below which ferritic steel breaks in a brittle rather than ductile manner. Radiation exposure from fast neutrons (> 1 mev) above about 10^{17} nvt may increase the NDT temperature of the vessel base metal. Extensive tests have established the magnitude of changes in the NDT temperature as a function of the integrated neutron exposure. The SAR presents pertinent test data for the type material (SA302B) used as the base metal for this vessel.

The initial NDT temperature of the main closure valves, the shell and head materials connecting to these flanges, and the connecting welds is 10°F. However, the vertical electroslag welds which terminate immediately below

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 34 TO
FACILITY OPERATING LICENSE NO. DPR-25

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION UNIT 3

DOCKET NO. 50-249

1.0 Introduction

By letter dated December 28, 1977, and supplement dated March 27, 1978 Commonwealth Edison (the licensee) proposed changes to the Technical Specifications of Facility Operating License DPR-25 for Dresden Nuclear Power Station Unit 3. The proposed changes relate to the replacement of 176 fuel assemblies constituting refueling of the core for sixth cycle operation at power levels up to 2527 MWt (100% power).

In support of the reload application the licensee has provided the GE BWR Reload 5 licensing submittal for Dresden Unit 3 (Reference 1), proposed Technical Specification changes (Reference 2), a Loss of Coolant Accident (LOCA) analysis report (Reference 3) and responses to NRC requests for additional information (Reference 4).

The information presented in the licensing submittal closely follows the guidelines of Appendix A of NEDO-20360 (Reference 5). Although later supplements to this report are undergoing review by the staff, portions of this topical have been found applicable for reactors containing 8x8 reload fuel and are acceptable to the staff when supplemented with information required by our status report (Reference 6). The supplemental information provided by the licensee and the staff's evaluation thereof are summarized below.

2.0 Evaluation

2.1 Nuclear Characteristics

For Cycle 6 operation of Dresden Unit 3, 156 fresh 8x8 fuel bundles (8D262) with an enrichment of 2.62% U235 by weight and 20 fresh 8x8 fuel bundles (8D250) with an enrichment of 2.50% U235 by weight will be loaded into the core. In addition, 164 7x7 assemblies from Cycle 1, 52 7D230 assemblies from Cycle 2, 44 8D250 assemblies from Cycle 3, 108 8D250 and 32 8D262 assemblies from Cycle 4, and 60 8D250 and 88 8D262 assemblies from Cycle 5 will remain in the core. Thus, for Cycle 6 approximately 24% of the 724 fuel bundles will be fresh fuel.

As indicated by the loading diagram presented in Reference 1, the fresh fuel will be distributed symmetrically throughout the core.

The nuclear characteristics of the reload 8D250 and 8D262 fuel bundles are identical to those previously loaded in the core. The licensee therefore states that the total control system worth, and the temperature and void dependent behavior of the reconstituted core will not differ significantly from those values previously reported for Dresden Unit 3.

The shutdown margin of the reconstituted core meets the Technical Specification requirement that the core be at least 0.25% Δk subcritical in the most reactive condition throughout the cycle when the highest worth control rod is fully withdrawn and all other rods are fully inserted. For Cycle 6 the licensee has calculated the minimum shutdown margin to be 0.014 Δk. The effect of settling of B_4C in inverted poison tubes in control rods will not have a significant effect on the Cycle 6 shutdown margin.

The information presented in Reference 1 indicates that a boron concentration of 600 ppm in the moderator will bring the reactor subcritical by at least 0.033 Δk at 20°C, xenon free. Therefore, the alternate shutdown requirement of the General Design Criteria is met by the Standby Liquid Control System.

The effective multiplication factor, k_{eff}, of the 8x8 fuel will be equal to or less than 0.90 for normal storage conditions. This is achieved since the peak uncontrolled k_∞ of all 8x8 fuel bundles, within the applicable exposure and temperature range, is less than 1.30 at 65°C. Therefore, the Technical Specification requirements for the storage of fuel at Dresden Unit 3 are met (Reference 5).

The full power scram reactivity curves used for the Reload 5 cycle are shown in Figure 6.6 of Reference 1. The scram curves used in the anticipated transient analyses include a design conservatism factor of 0.8 which is acceptable to the staff (Reference 6).

Based on our review of the information presented in the Dresden Unit 3 licensing submittal (Reference 1) as supplemented by applicable portions of the generic 8x8 reload report (Reference 5) and the staff's acceptance thereof (Reference 6), we have determined that the nuclear characteristics and expected performance of the reconstituted core for Cycle 6 are acceptable.

2.2 Mechanical Design

The Reload 5 fuel has the same mechanical configuration and fuel bundle enrichments as the 8D250 and 8D262 assemblies described in the 8x8 generic reload report (Reference 5). The finger springs and the improved water rod design described in Section 3 of Reference 5 have been adopted.

The generic 8x8 reload report (Reference 5), supplements of which are under review, has been found acceptable for use for reactors containing 8x8 reload fuel, when supplemented with information required by our status report (Reference 6) on the GE generic report evaluation. On the basis of our review of the generic 8x8 reload report and the reload submittal we conclude that the mechanical design of the Dresden Unit 3 Reload 5 fuel is acceptable.

2.3 Thermal-Hydraulics

The GE generic 8x8 fuel reload topical report (Reference 5) and GETAB (Reference 7) are referenced to provide the description of the thermal-hydraulic methods which were used to calculate the thermal margins. Application of the GETAB establishes:

(1) the fuel damage safety limit Minimum Critical Power Ratio (MCPR),

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

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

We have evaluated the Dresden Unit 3 Cycle 6 thermal margins based on the GETAB report and plant specific input information provided by the licensee. The staff evaluation of these margins is reported in the following subsections.

2.3.1 Fuel Cladding Integrity Safety Limit MCPR

The fuel cladding safety limit MCPR of 1.06 has been established, based on the GETAB (Reference 7) statistical analysis, to assure that 99.9% of the fuel rods in the core will not experience boiling transition during abnormal operational transients (Reference 8). This limit is applied for both core-wide and localized transients or perturbations to the expected CPR distribution.

The uncertainties in core and system operating parameters and the GEXL correlation uncertainties expected for Cycle 6 operation of Dresden Unit 3 are the same as those used for the original statistical analysis (Table 4-2 of Reference 5) on which the fuel cladding safety limit MCPR is based. The bundle power distribution for Cycle 6 is expected to include fewer high power bundles than the distribution assumed for the original statistical analysis as is indicated by comparing Figure 4-3 with Figures 4-4.1 through 4-4.4 of Reference 5. Therefore, it is conservative to apply the fuel cladding safety limit MCPR of 1.06 to Cycle 6 operation of Dresden Unit 3.

2.3.2 Operating Limit MCPR

Various transients or perturbations to the CPR distribution could reduce the MCPR below the intended operating limit during Cycle 6 operation of Dresden Unit 3. The most limiting operational transients and the fuel loading error have been analyzed by the licensee to determine which could potentially induce the largest reduction in MCPR.

The transients evaluated were the generator load rejection without bypass, the turbine trip with failure of the bypass valves, loss of 145°F of feedwater heating, and the control rod withdrawal error. Initial conditions and transient input parameters as specified in Table 4-2, Table 6-1 and Figure 6-6 of Reference 1 were assumed.

The input to the transient calculations and the application of the analysis methods of Reference 5 have been reviewed and determined to provide appropriate conservatism for determination of the operating limit MCPR for Dresden Unit 3 during Cycle 6.

The calculated reductions in CPR during each of the operational transients have been tabulated in Reference 1. For the 7x7 fuel the ΔCPR for the rod withdrawal error is the largest. For the 8x8 fuel

the ΔCPR for the generator load rejection is the largest. Addition of these ΔCPR's to the safety limit MCPR of 1.06 would give the operating limit MCPR's for each fuel type which would protect against boiling transition during plant transients. On this basis the licensee has calculated an operating limit MCPR of 1.29 for both the 7x7 and 8x8 fuel.

The licensee has also analyzed fuel loading errors (References 1 and 4). The worst error is one in which a fresh 8D262 bundle is placed in an exposed 8D250 bundle location. Should this occur, an even higher operating limit MCPR for the 8x8 fuel would be required to ensure that for this localized perturbation the CPR at the misloading site would not be below the safety limit of 1.06 during steady state operation and that the fuel rods in the misloaded fuel assembly would not experience transition boiling. On this basis the licensee has calculated that an operating limit MCPR of 1.36 for the 8x8 fuel is sufficient so that in the event of a fuel loading error the MCPR will not be below the 1.06 safety limit during steady state operation.

However, the Technical Specifications for Dresden Unit 3 require that the operating limit MCPR's based on the most limiting operational transient analysis be increased by 0.10 to account for a possible fuel loading error. This results in a Technical Specification operating limit MCPR of 1.39 for both the 7x7 and 8x8 fuel. Should there be no fuel loading error, then with this proposed operating limit MCPR, 99.9% of the fuel rods will avoid transition boiling by an extra margin during any operational transient.

We have reviewed the analyses performed to establish the operating limit MCPR and find the value of 1.39 acceptable.

2.3.3 Operating MCPR Limits For Less Than Rated Power And Flow

For the limiting transient of recirculation pump speed control failure at lower than rated power and flow condition, the licensee will conform to the limiting conditions for operation stated in paragraph 3.5-K of the Technical Specifications. This requires that for core flows less than the rated flow, the licensee maintain the MCPR greater than the operating minimum value. The MCPR value for less than rated flow is the rated flow value of 1.39 multiplied by the appropriate K_f factor appearing in Figure 3.5-2 of the Technical Specifications. The K_f factor curves were generically derived and assure that the most lmiting 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 operational transients do not violate the thermal limits of the fuel or the pressure limits of the reactor coolant boundary.

2.4 Accident Analysis

2.4.1 ECCS Appendix K Analysis

On December 27, 1974, the Atomic Energy Commission issued an Order for Modification of License implementing the requirements of 10 CFR 50.46, "Acceptance Criteria and Emergency Core Cooling Systems for Light Water Nuclear Power Reactors." One of the requirements of the Order was that prior to any license amendment authorizing any core reloading"...the licensee shall submit a re-evaluation of ECCS cooling performance calculated in accordance with an acceptable evaluation model which conforms to the provisions of 10 CFR Part 50, §50.46." The Order also required that the evaluation shall be accompanied by such proposed changes in Technical Specifications or license amendments as may be necessary to implement the evaluation results and assumptions.

In December of 1976 the NRC staff was informed that certain input errors and computer code errors had been made in the evaluations that were provided under the requirements described above. An Order was issued to the Commonwealth Edison Company on March 11, 1977 (Reference 9), requiring that corrected, revised calculations fully conforming to the requirements of 10 CFR 50.46 be provided for Dresden Unit 3 as soon as possible. Such corrected analyses were provided for the present reload in Reference 3. The corrected analyses included correction of all input errors previously made and correction of all computer code errors. The corrected analyses were performed using a calculational model which contains several model changes approved by the NRC staff in a Safety Evaluation issued April 12, 1977 (Reference 10).

We have reviewed the corrected analyses submitted for the Reload in Reference 3. We conclude that the Dresden Unit 3 plant could be operated in accordance with the "MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE" values given in Table 5 of Reference 3. However, the licensee has not requested a change in the MAPLHGR limits at this time. Therefore, the MAPLHGR limits as given in Appendix A to the March 11, 1977 Order will remain in effect. We reviewed these limits and found that with one exception, the MAPLHGR limits given in the Order are less than or equal to the limits given in the new analyses.

For the one exception (8D250 fuel at 30,000 MWD/t) the MAPLHGR limit in the Order exceeds that of the new analyses by 0.1 KW/ft. However, the new analyses predict a corresponding PCT of only 2112°F. Based on sensitivity studies of PCT to increases in MAPLHGR, it is not expected that the PCT would increase by more than 20°F for the higher MAPLHGR limit. Therefore, margin to the 2200°F limit would still exist.

Also, the licensee has committed to submit either a request for a Technical Specification change which would incorporate the MAPLHGR limits given in Reference 3 or a reanalysis of the 8D250 fuel with a burnup of 30,000 MWD/t at the higher KW/ft limit. In either case, the licensee will make the submittal well before the 30,000 MWD/t burnup is reached. The staff finds this acceptable.

Therefore, we conclude that Dresden Unit 3, prior to reaching a burnup of 30,000 MWD/t, will be in conformance with all the requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.46 when: (1) it is operated in accordance with the "MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE" values given in Appendix A of Reference 9 and (2) when it is operated at a Minimum Critical Power Ratio (MCPR) equal to or greater than 1.20 (more restrictive MCPR limits are currently required for reasons not connected with the Loss-of-Coolant-Accident, as described elsewhere in this SER).

The corrected analyses presented in Reference 3 represent the "lead plant" analyses (i.e., complete break spectrum study) for BWR/3 plants. Our Safety Evaluation of December 2, 1977 supporting the Dresden Unit 2 reload for Cycle 6 (Reference 11) contains our detailed evaluation of these corrected analyses.

2.4.2 Main Steam Line Break Accident

Steam line break accidents which are postulated to occur inside containment are covered by the ECCS analysis discussed in section 2.4.1. The analysis of steam line break accidents occurring outside containment as presented by the licensee is acceptable based on our generic review of NEDO-20360 (References 5 and 6).

2.4.3 Fuel Loading Error

Fuel loading errors are discussed in References 1 and 4 for a fuel bundle placed in an improper location or rotated 180 degrees. For Dresden Unit 3 the worst potential fuel loading error for Cycle 6 was analyzed from an initial MCPR of 1.25. This resulted in a MCPR of 0.95 and a peak linear heat generation rate of 18.87 KW/ft. The initial MCPR must therefore be raised by at least 0.11 to a value of 1.36. However, it has been observed that initiating the analysis at a higher MCPR results in a greater ΔCPR. The licensee, therefore, has performed additional calculations which demonstrate that an initial MCPR equal to the operating limit of 1.39 will maintain the MCPR above the thermal safety limit of 1.06 in the event of a fuel loading error. The implications of the MCPR have been discussed previously. The peak LHGR associated with a fuel loading error is not large enough to cause fuel damage.

2.4.4 Control Rod Drop Accident

In Figures 6-1 through 6-3 of Reference 1 the licensee has shown that for Cycle 6 operation of Dresden Unit 3 the Doppler Coefficient as a function of fuel temperature is more negative and the reactivity insertion due to a dropped in-sequence control rod versus rod position is smaller than bounding curves of these quantities presented in Reference 5. Furthermore, the scram reactivity as a function of time at both 20°C and 286°C (Figure 6-4 and 6-5 of Reference 1) will be greater than the corresponding bounding function presented in Reference 5.

Based on the analysis presented in Reference 5 and the discussion above, it is concluded that no in-sequence rod drop accident will lead to peak fuel enthalpies greater than the 280 cal/gm design basis for Dresden Unit 3 Cycle 6.

2.4.5 Fuel Handling Accident

The fuel handling accident was addressed in the staff's Safety Evaluation Report (SER) on the FSAR dated November 18, 1970 (Reference 12) and in the staff's status report on the generic reload for 8x8 fuel (Reference 6). In the status report, the staff stated that the mechanical analysis of the fuel handling accident should be better justified. However, the staff concluded that the amount of fission products released from 8x8 assemblies in a refueling accident would not be significantly greater than from the 7x7 assemblies. This conclusion is not changed by the reload, and the conclusions of the SER (Reference 12) that the dose consequences of a fuel handling accident are appropriately within 10 CFR Part 100 guidelines are not changed.

2.5 Overpressure Analysis

The licensee has presented an analysis to demonstrate that during the most severe overpressure event an adequate margin (64 psi) exists between the peak vessel pressure and the ASME code allowable vessel pressure which is 110% of the vessel design pressure (Reference 1). The event analyzed was the closure of all main steam line isolation valves with indirect (high flux) scram.

The input to the calculations is listed in Table 6-1 of Reference 1 and includes both end of cycle and end of cycle minus 2000 MWD/t values of void coefficient, Doppler coefficient and scram characteristics.

The licensee referenced a sensitivity study (Reference 13) which demonstrates that should the transient be initiated at the maximum pressure permitted by the high pressure trip point rather than that assumed for the analysis there would be a reduction in the margin to the pressure limit of approximately 20 psi. It has also been shown that the increase in peak vessel pressure during an MSIV closure due to a failed safety valve would not reduce the margin to the limit by more than approximately 17 psi (Reference 14).

Furthermore, it has been demonstrated that should the MSIV transient be initiated at a value of reactor power slightly above the value assumed for the analysis (because of uncertainties in monitoring of power) there would not be a significant reduction in margin (approximately 10 psi at 102% power) (Reference 15).

Based on the analysis and the sensitivity studies submitted, the overpressure analysis for Dresden Unit 3 for Cycle 6 has been found acceptable.

2.6 Thermal Hydraulic Stability Analysis

The thermal hydraulic stability analyses and results are described in References 5 and 1, respectively. The results of the Cycle 6 analysis show that for both the 7x7 and 8x8 fuel the channel hydrodynamic stability, at either rated power and flow conditions or at the low end of the flow control range, is within the operational design guide in terms of decay ratio. Calculations were also performed by the licensee to assess the reactor power dynamic response at the two aforementioned reactor operating conditions. The results of this analysis showed that the reactor core decay ratios at both conditions are well within the operational design guide decay ratio. These results are acceptable to the NRC staff.

The NRC staff has expressed generic concerns regarding the least stable reactor condition allowed by Technical Specifications. This condition could be reached during an operational transient from high power where the plant sustains a trip of both recirculation pumps. The concerns are motivated by increasing decay ratios as equilibrium fuel cycles are approached and as fuel designs change. The staff concerns relate to both the consequences of operating at an ultimate decay ratio and the capacity of analytical methods to accurately predict decay ratios. The General Electric Company is addressing the staff concerns through meetings, topical reports and a test program.

Until this issue has been resolved generically, the staff has imposed a requirement on Dresden Unit 3 which will restrict plant operations in the natural circulation flow mode. The licensee has agreed to this Technical Specification limitation. The restriction will provide a significant increase in the reactor core stability margins during Cycle 6. On the basis of the foregoing, the NRC staff considers the thermal-hydraulic stability of Dresden Unit 3 to be acceptable.

2.7 Recirculation Pump Startup From The Natural Circulation Operational Mode

During a recent BWR reload review (Reference 16), the question of recirculation pump startup from the natural circulation operational mode was raised. This pump startup could increase flow, collapse moderator voids, and subsequently result in a reactivity insertion transient. The consequences of such an accident sequence have not been previously evaluated. Therefore, authorization to operate in this fashion would

require additional analyses as to this accident sequence and its consequences. In the absence of this information, the Technical Specifications have been amended to prohibit operation of the reactor with natural recirculation flow when the reactor is operating at greater than 25% full rated power. The consequences of startup of a recirculation pump at below 25% reactor full power operation have been analyzed by the staff and have been found to be acceptable.

3.0 Physics Startup Testing

The licensee will perform a series of physics startup tests to provide assurance that the conditions assumed for the transient and accident analysis calculations will be met during Cycle 6. The tests, in addition to verifying the core loading and minimum shutdown margin, will assure that the incore monitoring system is functioning as expected and that the process computer has been reprogrammed to properly reflect changes associated with the reload.

The objectives, methods, and criteria for these tests have been described in References 4 and 17. The results of the tests will be submitted to the NRC within 45 days of startup.

The staff finds the licensee's plan for confirmatory testing and documentation acceptable.

4.0 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.

5.0 Conclusion

Based on our evaluation of the reload application and available information, we conclude that it is acceptable for the licensee to proceed with Cycle 6 operation of Dresden Unit 3 in the manner proposed.

We have reviewed the proposed changes to the Technical Specifications and find them acceptable.

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.

Dated: May 3, 1978

REFERENCES

1. "General Electric Boiling Water Reactor Reload 5 Licensing Submittal for Dresden Nuclear Power Station Unit 3," NEDO-24074, November 1977.

2. Letter from C. Reed, CECo, to E. G. Case, NRC, dated December 28, 1977.

3. Letter from R. L. Bolger, CECo, to E. G. Case, NRC, dated October 3, 1977 transmitting "Loss-of-Coolant Accident Analysis Report For Dresden Units 2, 3 and Quad Cities Units 1,2 Nuclear Power Stations (Lead Plant)", NEDO-24046, August 1977.

4. Letter from M. S. Turbak, CECo, to G. Lear, NRC, dated March 27, 1978.

5. "General Electric Boiling Water Reactor Generic Reload Application For 8x8 Fuel," NEDO-20360, Rev. 1, Supp. 4, April 1, 1976.

6. Status Report on the Licensing Topical Report, "General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel," NEDO-20360, Revision 1 and Supplement 1 by Division of Technical Review, Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission, April 1975.

7. "General Electric Thermal Analysis Basis (GETAB): Data Correlation and Design Application," NEDO-10958, November 1973.

8. Letter from J. A. Hinds, GE, to W. Butler, AEC, transmitting Responses to the Third Set of AEC Questions on the General Electric Licensing Topical Reports, NEDO-10958 and NEDE-10958, dated July 24, 1974.

9. Letter from D. L. Ziemann, NRC, to R. L. Bolger, CECo, transmitting Order for Modification of License, dated March 11, 1977.

10. Letter from K. R. Goller, NRC, to G. G. Sherwood, GE, transmitting "Safety Evaluation for General Electric ECCS Evaluation Model Modifications," dated April 12, 1977.

11. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 33 to Provisional Operating License No. DPR-19, Commonwealth Edison Company, Dresden Nuclear Power Station Unit 2, Docket No. 50-237, dated December 2, 1977.

12. Safety Evaluation by the Division of Reactor Licensing, U. S. Atomic Energy Commission, in the Matter of Commonwealth Edison Company, Dresden Nuclear Power Station Unit 3, Docket No. 50-249, dated November 18, 1970.

13. Letter from M. S. Turbak, CECo, to D. K. Davis, NRC, dated April 25, 1977.

14. Letter from I. F. Stuart, GE, to V. Stello, NRC, dated December 23, 1975.

15. Letter from R. L. Gridley, GE, to D. G. Eisenhut, NRC, dated September 12, 1977.

16. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 33 to License No. DPR-49, Duane Arnold Energy Center (Docket No. 50-331), dated May 6, 1977.

17. Letter from M. S. Turbak, CECo, to D. K. Davis, NRC, dated November 21, 1977.

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. 34 to Facility Operating License No. DPR-25, issued to Commonwealth Edison Company (the licensee), which revised the license and Technical Specifications appended thereto for operation of Dresden 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, (2) incorporated revised MCPR limits in response to the plant specific analysis for reload 5, and (3) modified License Condition 3.E to more concisely state plant operating plan described in the licensee reload submittal dated December 28, 1977.

The application for the amendment 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 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 December 28, 1977 and supplement dated March 27, 1978, (2) Amendment No. 34 to License No. DPR-25 and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Morris Public Library, 604 Liberty Street, Morris, Illinois 60451. A single copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland this 3rd day of May 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors