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Docket No. 50-237 DEC 2 1977

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

Gentlemen:

In response to your request dated September 12, 1977, and two supplements thereto, both dated November 21, 1977, the Commission has issued the enclosed Amendment No. 33 to Facility Operating License No. DPR-19 for Unit No. 2 of the Dresden Nuclear Power Station.

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.F to more concisely state end-of-cycle scram reactivity conditions for core Reload No. 3.

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

Sincerely,

Original Signed by
Don K. Davis

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment No. 33 to License No. DPR-19
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

Notified B. Lee at CECo (in response to his call) 12/2/77 ~5:30pm and Dick Knop at Reg III of issuance of this package

DD changes on pp 5, 6, 13 subject to 4 copy issuance as per conversation

OFFICE	ORB#2	ORB#2	OELD	ORB#2	
SURNAME	RDiggs	RBevan:acr	D SWANSON	DDavis	
DATE	12/ 2 /77	12/ 2 /77	12/ 2 /77	12/ 2 /77	

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

December 2, 1977

Docket No. 50-237

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

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Sincerely,

Paul W. O'Connor
for
Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment No. 33 to License No. DPR-19
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

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Springfield, Illinois 62761

Mr. William Waters
Chairman, Board of Supervisors of Grundy County
Grundy County Courthouse
Morris, Illinois 60450

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Federal Activities Branch
Region V Office
ATTN: EIS COORDINATOR
230 South Dearborn Street
Chicago, Illinois 60604

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 33
License No. DPR-19

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

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated September 12, 1977, as supplemented on November 21, 1977, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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

   3.B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 33, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   3.F. 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 Reload No. 3 licensing submittal for Dresden Unit 2 (NEDO-24034). 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

Paul W O'Connor

for, Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 2, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 33

FACILITY OPERATING LICENSE NO. DPR-19

DOCKET NO. 50-237

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
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 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.

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(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

Table 3.2.3

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)	$\leq [.65W + 40]$ (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

Amendment No. 21, 33

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 2	
1.39	(7 x 7 fuel)
1.37	(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.

81D

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 or generator 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 each fuel type.

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. Freasthe, October 1, 1973 (Proprietary).

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 limitations.

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

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

1.0 Introduction

By letters dated September 12, 1977 and November 21, 1977, Commonwealth Edison (the licensee) proposed changes to the Technical Specifications of Provisional Operating License DPR-19 for Dresden Nuclear Power Station Unit 2. The proposed changes relate to the replacement of 192 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 3 licensing submittal for Dresden Unit 2 (Reference 1), proposed Technical Specification changes (References 2 and 5), 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 6). 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 7). 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 2, 192 fresh 8x8 fuel bundles (8D250) with an enrichment of 2.50% U235 by weight will be loaded into the core. In addition, 216 7x7 assemblies from Cycles 1, 2 and 3, 32 7D230 and 124 8D250 assemblies from Reload 1 for Cycle 4, and 80 8D250 and 80 8D262 assemblies from Reload 2 for Cycle 5 will remain in the core. Thus, for Cycle 6 approximately 27% of the 724 fuel bundles will be fresh fuel.

As indicated by the loading diagram presented in Reference 5, the fresh fuel will be distributed symmetrically throughout the core.

The nuclear characteristics of the reload 8D250 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 2.

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. This occurs at the beginning of cycle. 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 0.035 Δk at 20 degrees 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 degrees C. Therefore, the Technical Specification requirements for the storage of fuel at Dresden Unit 2 are met (Reference 6).

The full power scram reactivity curves used for the Reload 3 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 7).

Based on our review of the information presented in the Dresden Unit 2 licensing submittal (Reference 1) as supplemented by applicable portions of the generic 8x8 reload report (Reference 6) and the staff's acceptance thereof (Reference 7), 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 3 fuel has the same mechanical configuration and fuel bundle enrichments as the 8D250 assemblies described in the 8x8 generic reload report (Reference 6). The finger springs and the improved water rod design described in Section 3 of Reference 6 have been adopted.

The generic 8x8 reload report (Reference 6), 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 7) 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 2 Reload 3 fuel is acceptable.

2.3 Thermal-Hydraulics

The GE generic 8x8 fuel reload topical report (Reference 6) and GETAB (Reference 8) 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 2 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 8) statistical analysis, to assure that 99.9% of the fuel rods in the core will not experience boiling transition during abnormal operational transients (Reference 9). 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 2 are the same as those used for the original statistical analysis (Table 4-2 of Reference 6) 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 6. Therefore, it is conservative to apply the fuel cladding safety limit MCPR of 1.06 to Cycle 6 operation of Dresden Unit 2.

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 2. 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 degrees 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 6 have been reviewed and determined to provide appropriate conservatism for determination of the operating limit MCPR for Dresden Unit 2 during Cycle 6.

The calculated reductions in CPR during each of the operational transients have been tabulated in References 1 and 5. 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. The licensee has also analyzed fuel loading errors. The worst error is one in which a fresh 8x8 bundle is placed in an exposed 8x8 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 1.37 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. Furthermore, should there be no fuel loading error, then with the proposed operating limit MCPR, 99.9% of the fuel rods will avoid transition boiling by an extra margin during any operational transient. The licensee has calculated an operating limit MCPR of 1.39 for the 7x7 fuel based on the results of the rod withdrawal error analysis. The staff has reviewed the operating limit MCPR values for the present operating cycle, and finds the values adopted to be 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 values. The MCPR values for less than rated flow are the rated flow values of 1.37 and 1.39 multiplied by the respective K_f factors appearing in Figure 3.5-2 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 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 10), requiring that corrected, revised calculations fully conforming to the requirements of 10 CFR 50.46 be provided for Dresden Unit 2 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 11).

We have reviewed the corrected analyses submitted for the Reload in Reference 3. We conclude that the Dresden Unit 2 plant will be in conformance with all 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 10 (the Commission's Order for Modification of License dated March 11, 1977) 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. The analyses provide all the information requested in the NRC letter to GE on June 30, 1977 (Reference 12) regarding number of breaks to be analyzed, documentation to be provided, etc. However, in order to assure that all BWR/3 plants identify and analyze their most limiting break size and location, each plant which references these analyses must consider certain features of the analyses. A description of these particular features is provided.

The break spectrum (i.e., peak clad temperature vs. break size) shows that the particular break producing the highest clad temperature (PCT) for the lead plants (Dresden 2/3 and Quad Cities 1/2) is the complete severence of a recirculation pump suction line. This break is herein called the limiting break. Reasons why the analysis of this break produced the highest PCT for D 2/3 and QC 1/2 are presented below.

The suction line break location is limiting for BWR/3 plants since it is the largest liquid line in the primary system.* The largest liquid line break results in the earliest core flow decay and consequent loss of nucleate boiling capability. It also results in the most rapid loss of primary system water and consequent uncovery of the high power plane. Both of these effects reduce the amount of stored heat that can be eliminated from the fuel before the fuel ceases to be covered by water and loses its ability to transfer heat. The fuel therefore enters its uncovered (assumed adiabatic heatup) period with a larger heat content which ultimately contributes to a higher PCT for large breaks at this location.

The only remaining factor that strongly influences PCT is the duration of the uncovered period. Duration of the uncovered period is a rather complex function of plant geometry, depressurization rate, counter current flow limiting (CCFL) phenomena, and available ECCS. If the longest uncovery time period is predicted for the largest possible size break at the limiting location, then all significant phenomena act together to cause that break to produce the highest PCT. If the longest uncovery time does not exist for the biggest break then it is necessary to more closely examine the interaction of the three effects versus break size. By examining these effects (loss of nucleate boiling time, core uncovery time and duration of uncovered period) the break size at the limiting location that produces the highest PCT can be determined.

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*LPCI Modified BWR/4 plants have a different limiting break location. With the most limiting single failure, these plants have at least one LPCI system available for the suction (largest) break but no LPCI flow for the discharge (smaller) break. This difference more than compensates for the larger size of the suction break and makes the smaller discharge break become limiting (Reference 13). However, BWR/3 ECCS equipment is arranged so that a single failure can prevent all LPCI flow for either break location, so the "compensating effects" situation does not exist and the larger pipe break location is clearly limiting.

In the case of Dresden 2/3 and Quad Cities 1/2, break sizes approximately 35% and approximately 60% of the largest suction line break area are predicted to remain uncovered longer than the largest suction line break. Plant specific calculations were performed (for D 2/3 and Q/C 1/2) for both of these break sizes and for the largest (100%) break size at the limiting location. The results demonstrated that the net effect of the interaction of the three phenomena described above results in the highest PCT for the largest suction line break (100%) for Dresden 2/3 and Quad Cities 1/2. Therefore, the MAPLHGR limits given in Tables 5B, C, E and F of Reference 3 could be applied to Dresden Unit 2 since they were developed by analysis of the largest break size at the limiting location.

However, it should be noted that the analysis of the "35% break" resulted in a PCT that was only 19 degrees F lower than the "100% break" PCT, and the "60% break" analysis resulted in a PCT that was only 41 degrees F lower than the "100% break" PCT.

Other BWR/3 plants will reference these "lead plant" analyses. However, slight differences in plant geometry could cause sufficiently long uncovery intervals for certain smaller breaks such that one of the smaller breaks could become limiting for those other plants. Therefore, the reasons why smaller breaks may have longer uncovery times than the largest break for certain BWR/3's and must be considered by non-lead plants are presented below.

1) The new "corrected" ECCS model predicts continued depressurization later in the transients. This maximizes steam generation and CCFL effects, thereby maximizing ultimate reflood delay and the uncovery time interval. The break flow area, primary system volume, and internal geometry (bypass flow area and peripheral bypass area) vary from plant to plant. This tends to vary the particular break size where the above effects are most pronounced when comparing PCT vs. break size results among several plants. However, the general trend is for this effect to be more pronounced for break sizes somewhat below the maximum possible size due to the following reason. For the largest break, even with the new model, depressurization (and resulting steam generation and CCFL effects) tends to be largely complete before spray initiation. That is, at the time when the spray flow is trying to go down through the core, most of the steam generation has already occurred. Therefore, the spray is relatively less impeded by CCFL effects on the largest breaks and reflood tends

to be less delayed. However, for somewhat smaller breaks, significant steam generation is still occurring when spray flow is trying to penetrate the core. This effect tends to maximize CCFL effects and reflood delays (and therefore uncovery interval) in the range of break sizes somewhat below the largest break area.

2) During core blowdown, water level calculations are performed by the SAFE code until: 1) the core sprays are on, and 2) the water level as predicted by SAFE is below the upper fuel tie plate. When both of these conditions are met, water level calculations are shifted to the REFLOOD code which contains the CCFL effects model. This code shift allows a more conservative calculation of water level after the two quoted conditions are met, i.e., when CCFL effects may exist and must be considered. However, there are two water level calculational methods programmed into the REFLOOD code, both acceptably conservative but different. The particular method used depends on core conditions when the calculation shifts to REFLOOD. For a given plant and break size, one or the other method will be used. However, analysis of a slightly different break size for the same plant may result in use of the other method. This shift in methods between adjacent break size analyses may cause (or at least amplify) sudden changes in the core uncovery interval vs. break size curve. When the shift from SAFE to REFLOOD is made, Small Break Methods (SBM)* are used in REFLOOD if the water level is in the active core region, and Large Break Methods (LBM)* are used if the water level is below the active core region.

   The two most significant differences between the small and large break methods in REFLOOD are:

   a) Use of the Vaporization Correlation: The vaporization of spray water in the core during the period when core sprays are operating is calculated using a bounding correlation. The correlation requires as input the PCT at the time of spray initiation.

*This is not to be confused with use of the Small Break model, which implies use of Small Break Methods in REFLOOD but also implies other changes such as use of a non-CHASTE-code heatup analysis. These other changes (other than Small Break Methods Within REFLOOD) are not implied here.

The LBM uses a constant value of PCT during the transient whereas the SBM uses a continuously increasing value. Both methods have been accepted by the staff. This difference generally results in a more conservative calculation of the reflooding time using the SBM.

b) Level and Vaporization Following Bottom Reflooding: The LBM uses an empirically based void fraction of 0.50 for calculating the swollen water level and the vaporization below the level. The SBM uses the conservative fuel rod heatup model with a reflooding heat transfer coefficient to calculate the level and the vaporization below the level. This difference generally results in a more conservative calculation of the reflooding time using the SBM.

For a given plant and break size, the selection of the SBM or LBM in REFLOOD (using the criteria described above) depends upon primary system volume to break area ratio, mass loss, depressurization rate, ECCS initiation time, etc. For Dresden 2/3 and Quad Cities 1/2, these factors cause the LBM to be used for the largest break areas (above 60%) and the SBM to be used for the smallest break areas (below 40%). Between 40% and 60%, the method used changes several times.

As previously indicated, these shifts in method can cause (or at least amplify) peaks in the core uncovery time interval vs. break size curve, and hence in the PCT vs. break size curve. Although smaller breaks do not become limiting for the Dresden 2/3 and Quad Cities 1/2 analyses, slight differences in other plants referencing these analyses could cause a smaller break to be limiting.

The effects described above must therefore be considered by other plants referencing these "lead plant" analyses, to assure that the most limiting break is identified for each plant.

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 6 and 7).

2.4.3 Fuel Loading Error

Fuel loading errors are discussed in References 1, 4, and 5 for a fuel bundle placed in an improper location or rotated 180 degrees. For Dresden Unit 2 the worst potential fuel loading error for Cycle 6 was analyzed from an initial MCPR of 1.26. This resulted in a MCPR of 0.98 and a peak linear heat generation rate of 17.8 KW/ft. The initial MCPR must therefore be raised by at least 0.08 to a value of 1.34. However, it has been observed that initiating the analysis at a higher MCPR results in a greater ΔCPR. The licensee has conservatively estimated the increase in ΔCPR to be 0.03. The plant operating MCPR for 8x8 fuel will therefore be set at 1.37. The licensee will confirm the appropriateness of this value by an analysis to be submitted in the near future. The staff finds this acceptable based on the sensitivity of ΔCPR to initial MCPR seen to date. 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 2 the Doppler Coefficient as a function of fuel temperature and the reactivity insertion due to a dropped in-sequence control rod versus rod position are smaller than bounding curves of these quantities presented in Reference 6. Furthermore, the scram reactivity as a function of time at 286 degrees C (Figure 6-5 of Reference 1) will be greater than the corresponding bounding function presented in Reference 6. At 20 degrees C the specific scram reactivity function (Figure 6-4 of Reference 1) is slightly less than the corresponding bounding analysis curve beyond 3.75 seconds. However, sufficient negative reactivity will be added in the first 3.75 seconds to compensate for the positive reactivity added by the dropped rod.

Based on the analysis presented in Reference 6 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 2 Cycle 6.

2.4.5 Fuel Handling Accident

With respect to fuel handling accidents, the licensee noted that the description and analyses of this event provided in the FSAR and discussed in the generic 8x8 reload report (Reference 6) are applicable to this reload. That is, the total activity released to the environment and the

radiological exposures for the 8x8 fuel will be less than those values presented in the FSAR for the 7x7 core. As identified in the FSAR the radiological exposures for this accident with 7x7 fuel are well below the guidelines set forth in 10 CFR 100. Therefore, it is concluded that the consequences of this accident for the 8x8 fuel will also be well below the 10 CFR 100 guidelines.

2.5 Overpressure Analysis

The licensee has presented an analysis to demonstrate that during the most severe overpressure event an adequate margin (61 psi) exists between the peak vessel pressure and the AMSE 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 1500 $MWD/_t$ values of void coefficient, Doppler coefficient and scram characteristics.

The licensee referenced a sensitivity study (Reference 14) 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 15 psi (Reference 15).

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 16).

Based on the analysis and the sensitivity studies submitted, the overpressure analysis for Dresden Unit 2 for Cycle 6 has been found acceptable.

2.6 Thermal Hydraulic Stability Analysis

The thermal hydraulic stability analyses and results are described in References 6 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 improve. 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 2 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 2 to be acceptable.

2.7 Recirculation Pump Startup From The Natural Circulation Operational Mode

During a recent BWR reload review (Reference 17), 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 at 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 conduct physics startup tests which in addition to verifying the predicted shutdown margin, will provide assurance that the incore monitoring instrumentation is functioning properly, that

the process computer is programmed correctly, and that the core is loaded as intended. These tests will provide additional assurance that the Cycle 6 core as loaded is consistent with the physics input to the transient and accident analyses contained in the reload licensing submittal (Reference 1). The results of the tests will be submitted to the NRC within 90 days of startup.

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

4.0 Conclusions

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 2 in the manner proposed.

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

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.

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 change 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: December 2, 1977

REFERENCES

1. "General Electric Boiling Water Reactor Reload 3 Licensing Submittal for Dresden Nuclear Power Station Unit 2," NEDO-24034, June 1977.

2. Letter from R. L. Bolger, CECo, to E. G. Case, NRC, dated September 12, 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 D. K. Davis, NRC, dated November 21, 1977.

5. Letter from R. L. Bolger, CECo, to E. G. Case, NRC, dated November 21, 1977.

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

7. 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.

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

9. 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.

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

11. 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.

12. Letter from D. G. Eisenhut, NRC, to E. D. Fuller, GE, "Documentation of the Reanalysis Results for the Loss-of-Coolant Accident (LOCA) of Lead and Non-Lead Plants," dated June 30, 1977.

13. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 30 to Facility Operating License No. DPR-59, James A. Fitzpatrick Atomic Power Station, dated September 16, 1977.

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

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

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

17. 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.

18. Letter from H. A. Zimmerman, GE, to M. S. Turbak, CECo, dated November 23, 1977.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-237

COMMONWEALTH EDISON COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 33 to Facility Operating License No. DPR-19, issued to Commonwealth Edison Company (the licensee), which revised the license and Technical Specifications appended thereto for operation of Dresden Station Unit No. 2 (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 3, and (3) modified License Condition 3.F to more concisely state end-of-cycle scram reactivity conditions for reload 3.

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 September 12, 1977 and two supplements thereto dated November 21, 1977, (2) Amendment No. 33 to License No. DPR-19 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 2nd day of December, 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Paul W. O'Connor

Paul W. O'Connor, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors