November 23, 1992

Docket No. 50-237

Mr. Thomas J. Kovach
Nuclear Licensing Manager
Commonwealth Edison Company-Suite 300
OPUS West III
1400 OPUS Place
Downers Grove, Illinois 60515

Dear Mr. Kovach:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. M84452)

The Commission has issued the enclosed Amendment No. 121 to Facility Operating License No. DPR-19 for Dresden, Unit 2. The amendment is in response to your application dated September 2, 1992.

The amendment will (1) incorporate, by reference, the new Siemens Nuclear Power's methodologies previously approved by the NRC staff, (2) increase the resultant Safety Limit Minimum Critical Power Ratio (SLMCPR) in the Technical Specifications, and (3) remove specifications referring to the SLMCPR for GE fuel.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original Signed By:

Byron L. Siegel, Project Manager
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 121 to DPR-19
2. Safety Evaluation

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cc:

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 121
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 2, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-19 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 121 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective prior to startup from the next refueling outage (Cycle 14).

FOR THE NUCLEAR REGULATORY COMMISSION

James E. Dyer, Director
Project Directorate III-2
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 23, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 121

FACILITY OPERATING LICENSE NO. DPR-19

DOCKET NO. 50-237

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

| REMOVE | INSERT |
|---|---|
| 1/2.1-1 | 1/2.1-1 |
| B 1/2.1-7 | B 1/2.1-7 |
| B 1/2.1-8 | B 1/2.1-8 |
| 6-19 | 6-19 |

## 1.1 SAFETY LIMIT

### FUEL CLADDING INTEGRITY

Applicability:

The Safety Limits established to preserve the fuel cladding integrity apply to these variables which monitor the fuel thermal behavior.

Objective:

The objective of the Safety Limits is to establish limits below which the integrity of the fuel cladding is preserved.

Specifications:

A. Reactor Pressure greater than 800 psig and Core Flow greater than 10% of Rated.

The existence of a minimum critical power ratio (MCPR) less than 1.08 shall constitute violation of the MCPR fuel cladding integrity safety limit.

When in Single Loop Operation, the MCPR safety limit shall be increased by 0.01.

## 2.1 LIMITING SAFETY SYSTEM SETTING

### FUEL CLADDING INTEGRITY

Applicability:

The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity Safety Limits from being exceeded.

Objective:

The objective of the Limiting Safety System Settings is to define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity Safety Limits from being exceeded.

Specifications:

A. Neutron Flux Trip Settings
The limiting safety system trip settings shall be as specified below:

1. APRM Flux Scram Trip Setting (Run Mode)
When the reactor mode switch is in the run position, the APRM flux scram setting shall be:

S less than or equal to $[.58W_D + 62]$ during Dual Loop Operation or S less than or equal to $[.58W_D + 58.5]$ during Single Loop Operation

with a maximum setpoint of 120% for core flow equal to $98 \times 10^6$ lb/hr and greater, where:

s - setting in percent of rated thermal power.

1.1 SAFETY LIMIT BASES (Cont'd.)

power ratio (CPR) which is the ratio of the bundle power which would produce the onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the Minimum Critical Power Ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setpoints via the instrumented variables. (Figure 2.1-3).

The MCPR Fuel Cladding Integrity Safety Limit assures sufficient conservatism in the operating MCPR limit that in the event of an anticipated operational occurrence from the limiting condition for operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (MCPR=1.00) and the MCPR Fuel Cladding Integrity Safety Limit is based on a detailed statistical procedure which considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the safety limit is the uncertainty inherent in the NRC-approved critical power correlation. Refer to Specification 6.6.A.4 for the methodology used in determining the MCPR Fuel Cladding Integrity Safety Limit.

The NRC-approved critical power correlation is based on a significant body of practical test data, providing a high degree of assurance that the critical power as evaluated by the correlation is within a small percentage of the actual critical power being estimated. The assumed reactor conditions used in defining the safety limit introduce conservatism into the limit because boundingly high radial power peaking factors and boundingly flat local peaking distributions are used to estimate the number of rods in boiling transition. Still further conservatism is induced by the tendency of the NRC-approved correlation to overpredict the number of rods in boiling transition. These conservatisms and the inherent accuracy of the NRC-approved correlation provide a reasonable degree of assurance that during sustained operation at the MCPR Fuel Cladding Integrity Safety Limit there would be no transition boiling in the core. If boiling transition were to occur, however, there is reason to believe that the integrity of the fuel would not necessarily be compromised. Significant test data accumulated by the U.S. Nuclear Regulatory Commission and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative approach; much of the data indicates that LWR fuel can survive for an extended period in an environment of transition boiling.

During Single Loop Operation, the MCPR safety limit is increased by 0.01 to conservatively account for increased uncertainties in the core flow and TIP measurements.

1.1 SAFETY LIMIT BASES (Cont'd.)

If the reactor pressure should ever exceed the limit of applicability of the NRC-approved critical power correlation as defined in the NRC-approved methodology listed in Specification 6.6.A.4, it would be assumed that the MCPR Fuel Cladding Integrity Safety Limit had been violated. This applicability pressure limit is higher than the pressure safety limit specified in Specification 1.2.

Fuel design criteria have been established to provide protection against fuel centerline melting and 1% plastic cladding strain during transient overpower conditions throughout the life of the fuel. To demonstrate compliance with these criteria, fuel rod centerline temperatures are determined at 120% overpower conditions as a check against calculated centerline melt temperatures. FDLRC is incorporated to protect the above criteria at all power levels considering events which cause the reactor power to increase to 120% of rated thermal power.

B. Core Thermal Power Limit (Reactor Pressure less than 800 psia)

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flows will always be greater than 4.56 psi. Analyses show that with a flow of $28 \times 10^3$ lbs/hr. bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow

6.0 ADMINISTRATIVE CONTROLS (Cont'd.)

4. Core Operating Limits Report

a. Core operating limits shall be established and documented in the Core Operating Limits Report before each reload cycle or any remaining part of a reload cycle for the following:

1) The Control Rod Withdrawal Block Instrumentation for Table 3.2-3 of Specification 3.2.C.

2) The Average Planar Linear Heat Generation Rate (APLHGR) Limit and associated APLHGR multipliers for Specifications 3.5.I, 3.5.D.2, and 3.6.H.3.f.

3) The Local Steady State Linear Heat Generation Rate (LHGR) for Specification 3.5.J.

4) The Local Transient Linear Heat Generation Rate (LHGR) for Specification 3.5.K.

5) The Minimum Critical Power Operating Limit for Specification 3.5.L. This includes rated and off-rated flow conditions.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in the latest approved revision or supplement of the topical reports describing the methodology. For Dresden Unit 2, the topical reports are:

1) ANF-1125(P)(A), "Critical Power Correlation - ANFB."

2) ANF-524(P)(A), "ANF Critical Power Methodology for Boiling Water Reactors."

3) XN-NF-79-71(P)(A), "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors".

4) XN-NF-80-19(P)(A), "Exxon Nuclear Methodology for Boiling Water Reactors".

5) XN-NF-85-67(P)(A), "Generic Mechanical Design for Exxon Nuclear Jet Pump Boiling Water Reactors Reload Fuel".

6) XN-NF-81-22(P)(A), "Generic Statistical Uncertainty Analysis Methodology".

7) ANF-913(P)(A), "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses."

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 121 TO FACILITY OPERATING LICENSE NO. DPR-19

COMMONWEALTH EDISON COMPANY

DRESDEN NUCLEAR POWER STATION, UNIT 2

DOCKET NO. 50-237

## 1.0 INTRODUCTION

Siemens Nuclear Power (SNP) is currently utilized by Commonwealth Edison Company (CECo, the licensee) for the performance of the reload licensing calculations for Dresden, Units 2 and 3. Prior to Dresden, Unit 3, Cycle 13, SNP received NRC approval for the advanced methodology package for BWR reload design and safety analysis. By letter dated September 2, 1992, the licensee requested that the Dresden, Unit 2, Technical Specifications (TS) be modified to reflect the use of these NRC-approved methodologies for reload licensing calculations to determine the core operating limits at Dresden starting with Cycle 14 (which is currently scheduled for early April 1993). Specifically, the proposed amendment would incorporate new, NRC-approved SNP methodologies, increase the resultant Safety Limit Minimum Critical Power Ratio (SLMCPR) and remove TS referring to the SLMCPR for GE fuel.

## 2.0 EVALUATION

The NRC staff has approved the following SNP Topical Reports on reload licensing and safety analysis methodologies which Dresden, Unit 2, is proposing to use starting with Cycle 14.

ANF-1125(P)(A), "Critical Power Correlation - ANFB" - This ANFB correlation provides a generic tool for evaluating critical power and assessing thermal margin for all SNP BWR fuel designs. ANFB replaces the XN-3 calculation and will be used for both licensing and on-site core monitoring calculations and is applicable to all resident fuel types at Dresden, including those to be used for Unit 2, Cycle 14.

ANF-913 (P)(A), "COTRANSA2: A Computer Program for Boiling Water Reactor"

A system transient analysis code for BWRs which will be used by SNP to evaluate postulated limiting transients for future Dresden reloads starting with Cycle 14 for Unit 2. The code is based in part, on the previously approved X-COBRA and RELAX Codes.

ANF-524(P)(A), Revision 2, "ANF Critical Power Methodology for Boiling Water Reactors" - This is used to calculate a Maximum Critical Power Ratio (MCPR) Safety Limit that ensures 99.9% of the fuel rods avoid boiling transition. It includes the MCPR calculational procedure with the corresponding system and calculational uncertainties. The methodology also accounts for the effects of channel bow for single bundle lifetime channels.

XN-NF-80-19(P)(A), Volume 1, Supplement 3, "Advanced Nuclear Fuels Methodology for Boiling Water Reactors; Benchmark Results for the CASMO-3G/MICROBURN-B Calculational Methodology" - The CASMO-3G/MICROBURN-B Code is used by SNP for reload design, steady-state licensing, and plant core simulator support applications. It is multigroup transport theory calculation of the spatial flux and power distribution, cell multiplication, and isotopic depletion for two-dimensional BWR fuel assembly lattices and three-dimensional core simulation.

Consistent with NRC Generic Letter (GL) 88-16, the licensee is proposing that the first three of these approved topical reports, which are used to determine core operating limits, be incorporated by reference into Section 6.6.A.4.b of the Dresden, Unit 2, TS. The fourth topical report (XN-NF-80-19(P)(A)) is currently referenced in Section 6.6.A.4.b.4 of the TS.

The licensee has stated that the new SNP critical power methodology, which is based on the staff approved ANFB critical power correlation, accounts for the effects of channel bow for single bundle lifetime channels and is applicable for the Dresden, Unit 2, Cycle 14 reload since no second bundle lifetime channels are being used. The use of these new methodologies for Cycle 14 increases the Safety Limit MCPR (SLMCPR) from 1.05 to 1.08. This increase accounts for the effects of channel bow differences in core modeling (0.02 delta CPR increase) and for an additional conservatism (0.01 delta CPR) that has been included to accommodate minor changes in future reload designs to facilitate reload licensing under 10 CFR 50.59. As a result of this new methodology, the licensee has proposed an increase in the SLMCPR to 1.08 in Section 1.1.A of the Unit 2 TS. Also, the licensee has requested to remove the SLMCPR of 1.06 for GE fuel since it is not anticipated that GE fuel will be loaded into the Dresden, Unit 2, core in the near future.

The staff has reviewed the licensee's proposed TS changes and determined that they are acceptable. This conclusion is based on the following: this amendment is the same as the Dresden, Unit 3, amendment previously approved by the staff on August 5, 1991; the referenced methodologies have been previously approved by the staff; the same spectrum of limiting events for each reload will be used and analyzed under the new methodology; the increased SLMCPR adequately accounts for the potential effects of channel bow under the new methodology for Dresden, Unit 2, Cycle 14 with some additional conservatism; the new methodologies and SLMCPR increase will maintain the current margin of safety and fuel cladding integrity; and the removal of the SLMCPR of 1.06 for GE fuel is not a concern since GE fuel is not loaded in the core.

## 3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of this amendment. The State official had no comments.

## 4.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration, and there has been no public comment on such finding (57 FR 45079). Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

## 5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Byron Siegel

Date: November 23, 1992