January 31, 1990

Docket Nos. 50-454
and 50-455

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SSun

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

SUBJECT: AMENDMENT NO. 36 -USE OF VANTAGE 5 FUEL
(TAC NOs. 74166 AND 74167)

Dear Mr. Kovach:

The Commission has issued the enclosed Amendment No. 36 to Facility Operating License No. NPF-37 and Amendment No. 36 to Facility Operating License No. NPF-66 for the Byron Station, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated July 31, 1989.

These amendments approve changes to the Technical Specifications to allow the use of Vantage 5 fuel.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance is being filed with the Office of the <u>Federal Register</u> for publication.

Sincerely,

Leonard N. Olshan, Project Manager
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosures:
1. Amendment No. 36 to NPF-37
2. Amendment No. 36 to NPF-66
3. Safety Evaluation

cc w/enclosures:
See next page

PDIII-2:LA	PDIII-2:PM	PDIII-2:PD	OGC-OWFN
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

January 31, 1990

Docket Nos. 50-454
and 50-455

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

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IV, V and Special Projects

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2. Amendment No. 36 to NPF-66
3. Safety Evaluation

cc w/enclosures:
See next page

Mr. Thomas J. Kovach
Commonwealth Edison Company

Byron Station
Units 1 and 2

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-454

BYRON STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 36
License No. NPF-37

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated July 31, 1989, 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. NPF-37 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A as revised through Amendment No. 36 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John W. Craig

John W. Craig, Director
Project Directorate III-2
Division of Reactor Projects - III, IV, V and Special Projects

Attachment:
Changes to the Technical Specifications

Date of Issuance: January 31, 1990

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-455

BYRON STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 36
License No. NPF-66

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated July 31, 1989, 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 1;

   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. NPF-66 is hereby amended to read as follows:

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A (NUREG-1113), as revised through Amendment No. 36 and revised by Attachment 2 to NPF-60, and the Environmental Protection Plan contained in Appendix B, both of which were attached to License No. NPF-37, dated February 14, 1985, are hereby incorporated into this license. Attachment 2 contains a revision to Appendix A which is hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John W. Craig, Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 31, 1990

ATTACHMENT TO LICENSE AMENDMENT NOS. 36 AND 36

FACILITY OPERATING LICENSE NOS. NPF-37 AND NPF-66

DOCKET NOS. 50-454 AND 50-455

Revise Appendix A as follows:

Remove Pages	Insert Pages
2 - 8	2 - 8
B 2 - 1	B 2 - 1
B 2 - 2	B 2 - 2
3/4 1 - 4	3/4 1 - 4
3/4 1 - 5	3/4 1 - 5
3/4 1 - 19	3/4 1 - 19
3/4 2 - 4	3/4 2 - 4
3/4 2 - 7	3/4 2 - 7
3/4 2 - 8	3/4 2 - 8
B 3/4 1 - 2	B 3/4 1 - 2
B 3/4 2 - 1	B 3/4 2 - 1
B 3/4 2 - 4	B 3/4 2 - 4
B 3/4 2 - 5	B 3/4 2 - 5

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, $\tau_6 = 0$ s,
T'	$\leq$	588.4°F (Nominal T_{avg} at RATED THERMAL POWER),
K_3	=	0.00134,
P	=	Pressurizer pressure, psig,
P'	=	2235 psig (Nominal RCS operating pressure),
S	=	Laplace transform operator, s^{-1},

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant STARTUP tests such that:

(i) for $q_t - q_b$ between $-\infty$% and +10% (Unit 1 Cycle 3 and Unit 2 Cycle 2), and -32% and +13% (Unit 1 Cycle 4 and after; Unit 2 Cycle 3 and after), $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) for each percent that the magnitude of $q_t - q_b$ exceeds +10% (Unit 1 Cycle 3 and Unit 2 Cycle 2), and +13% (Unit 1 Cycle 4 and after; Unit 2 Cycle 3 and after), the ΔT Trip Setpoint shall be automatically reduced by 2.0% (Unit 1 Cycle 3 and Unit 2 Cycle 2), and 1.74% (Unit 1 Cycle 4 and after; Unit 2 Cycle 3 and after) of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds -32%, the ΔT Trip Setpoint shall be automatically reduced by 1.67% of its value at RATED THERMAL POWER (Unit 1 Cycle 4 and after; Unit 2 Cycle 3 and after)

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 3.9% of ΔT span.

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, and is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95 percent probability that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used (the WRB-1 correlation for Optimized Fuel Assembly (OFA) fuel and the WRB-2 correlation for VANTAGE 5 fuel in this application). The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the correlation DNBR limit (1.17 for both the WRB-1 and WRB-2 correlations).

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95 confidence that the minimum DNBR for the limiting rods is greater than or equal to the DNBR limit. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a design DNBR value which must be met in plant safety analysis using values of input parameters without uncertainties.

The design DNBR values are 1.34 and 1.32 for a typical cell and a thimble cell, respectively for OFA fuel, and 1.33 for a typical cell and 1.32 for a thimble cell for the VANTAGE 5 fuel. In addition, margin has been maintained in both designs by meeting safety analysis DNBR limits of 1.49 for a typical cell and 1.47 for a thimble cell for OFA fuel, and 1.67 and 1.65 for a typical cell and a thimble cell, respectively for the VANTAGE 5 fuel in performing safety analyses.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum design DNBR is no less than the design DNBR value, or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

REACTOR CORE (Continued)

These curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.49 for OFA fuel and 1.59 for VANTAGE 5 fuel. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.49\ [1 + 0.3\ (1-P)] \quad \text{for OFA fuel}$$

$$F^N_{\Delta H} = 1.59\ [1 + 0.3\ (1-P)] \quad \text{for VANTAGE 5 fuel}$$

Where P is the fraction of RATED THERMAL POWER.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the Setpoints to provide protection consistent with core Safety Limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System (RCS) from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor vessel, pressurizer, and the RCS piping, valves, and fittings are designed to Section III of the ASME Code for Nuclear Power Plants which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated Code requirements.

The entire RCS is hydrotested at 3110 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0 $\Delta k/k/^{\circ}F$ for the all rods withdrawn, hot zero THERMAL POWER condition, or

b. Less negative than -4.1×10^{-4} $\Delta k/k/^{\circ}F$ for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.1.3a. - MODES 1 and 2* only#.
Specification 3.1.1.3b. - MODES 1, 2, and 3 only#.

ACTION:

a. With the MTC more positive than the limit of Specification 3.1.1.3a. above, operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than 0 $\Delta k/k/^{\circ}F$ within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

3. A Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.

4. The provisions of Specification 3.0.4 are not applicable.

b. With the MTC more negative than the limit of Specification 3.1.1.3b. above, be in HOT SHUTDOWN within 12 hours.

*With K_{eff} greater than or equal to 1.

#See Special Test Exceptions Specification 3.10.3.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the predicted MTC to establish administrative rod withdrawal limits, as necessary, to assure that the limit of Specification 3.1.1.3a., above, is met throughout core life, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading, and

b. The MTC shall be measured at any THERMAL POWER and compared to -3.2×10^{-4} $\Delta k/k/^{\circ}F$ (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than -3.2×10^{-4} $\Delta k/k/^{\circ}F$, the MTC shall be remeasured, and compared to the EOL MTC limit of Specification 3.1.1.3b., at least once per 14 EFPD during the remainder of the fuel cycle.

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length shutdown and control rod drop time from the fully withdrawn position shall be less than or equal to 2.4 seconds (Unit 1 Cycle 3 and Unit 2 Cycle 2), and 2.7 seconds (Unit 1 Cycle 4 and after; Unit 2 Cycle 3 and after) from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 550°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the rod drop time of any full-length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop time within limits but determined with three reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to less than or equal to 66% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full-length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the Control Rod Drive System which could affect the drop time of those specific rods, and

c. At least once per 18 months.

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(Z)$

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q(Z)$ shall be limited by the following relationships:

$F_Q(Z) \leq [\frac{2.32}{P}]$ [K(Z)] for P > 0.5*, and

$F_Q(Z) \leq [4.64]$ [K(Z)] for P $\leq$ 0.5*.

$F_Q(Z) \leq [\frac{2.50}{P}]$ [K(Z)] for P > 0.5**,

$F_Q(Z) \leq [5.00]$ [K(Z)] for P $\leq$ 0.5**.

Where:

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}},$$

and K(Z) is the function obtained from Figure 3.2-2 for a given core height location.

APPLICABILITY: MODE 1.

ACTION:

With $F_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q(Z)$ exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% $F_Q(Z)$ exceeds the limit; and

b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced limit required by ACTION a., above; THERMAL POWER may then be increased provided $F_Q(Z)$ is demonstrated through incore mapping to be within its limit.

*Unit 1 Cycle 3 and Unit 2 Cycle 2

**Unit 1 Cycle 4 and after; Unit 2 Cycle 3 and after

SURVEILLANCE REQUIREMENTS (Continued)

2) When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) shall be within the limits provided in the OPERATING LIMITS REPORT for all core planes containing Bank "D" control rods and for all unrodded core planes;

f. The F_{xy} limits of Specification 4.2.2.2e., above, are not applicable in the following core planes regions as measured in percent of core height from the bottom of the fuel:

1) Lower core region from 0 to 15%, inclusive,

2) Upper core region from 85 to 100%, inclusive,

3) Within ± 2% of grid plane regions (except VANTAGE 5 assembly Intermediate Flow Mixer Grids) such that no more than 20% of the total core height in the center core region is affected, and

4) Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the Bank "D" control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L}, the effects of F_{xy} on $F_Q(Z)$ shall be evaluated to determine if $F_Q(Z)$ is within its limits.

4.2.2.3 When $F_Q(Z)$ is measured for other than F_{xy} determinations, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

3/4.2.3 RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 Indicated Reactor Coolant System (RCS) total flow rate and $F^N_{\Delta H}$ shall be maintained as follows for four loop operation.

a. RCS Total Flowrate $\geq$ 390,400 gpm, and

b. $F^N_{\Delta H} \leq 1.55 [1.0 + 0.3 (1.0-P)]$ for OFA fuel

$F^N_{\Delta H} \leq 1.65 [1.0 + 0.3 (1.0-P)]$ for VANTAGE 5 fuel

where:

Measured values of $F^N_{\Delta H}$ are obtained by using the movable incore detectors. An appropriate uncertainty of 4% (nominal) or greater shall then be applied to the measured value of $F^N_{\Delta H}$ before it is compared to the requirements, and

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

APPLICABILITY: MODE 1.

ACTION:

With RCS total flow rate or $F^N_{\Delta H}$ outside the region of acceptable operation:

a. Within 2 hours either:

1. Restore RCS total flow rate and $F^N_{\Delta H}$ to within the above limits, or

2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux-High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

MODERATOR TEMPERATURE COEFFICIENT (Continued)

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting MTC value -4.1×10^{-4} $\Delta k/k/°F$. The MTC value of -3.2×10^{-4} $\Delta k/k/°F$ represents a conservative value (with corrections for burnup and soluble boron) at a core condition of 300 ppm equilibrium boron concentration and is obtained by making these corrections to the limiting MTC value of -4.1×10^{-4} $\Delta k/k/°F$.

The Surveillance Requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC can be maintained within its limits. The BOL MTC measurement, combined with the predicted MTC throughout core life, will be used to impose administrative limits on rod withdrawal, as required during core life to ensure that MTC will always be less positive than 0 $\Delta K/K/°F$. This coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 550°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, (4) the reactor vessel is above its minimum RT_{NDT} temperature, and (5) the plant is above the cooldown steam dump permissive, P-12.

3/4.1.2 BORATION SYSTEMS

The Boron Injection System ensures that negative reactivity control is available during each MODE of facility operation. The components required to perform this function include: (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid transfer pumps, and (5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 350°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 1.3% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 15,780 gallons of 7000-ppm borated water from the boric acid storage tanks or 70,450 gallons of 2000-ppm borated water from the refueling water storage tank.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the minimum DNBR in the core greater than or equal to the appropriate DNBR limit (See Bases 2.1) during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods;

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power; and

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of the F_Q Limit times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full-length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

c. The control rod insertion limits of Specification 3.1.3.6 are maintained, and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided the Conditions a. through d. above are maintained. The combination of the RCS flow requirement (390,400 gpm) and the requirement on $F^N_{\Delta H}$ guarantee that the DNBR used in the safety analysis will be met.

Margin between the safety analysis limit DNBRs (1.49 and 1.47 for the OFA fuel typical and thimble cells, respectively and 1.67 and 1.65 for the VANTAGE 5 typical and thimble cells) and the design limit DNBRs (1.34 and 1.32 for the OFA fuel typical and thimble cells, and 1.33 and 1.32 for the VANTAGE 5 fuel typical and thimble cells, respectively) is maintained.

A fraction of this margin is utilized to accommodate the transition core DNBR penalty (maximum of 12.5%) and the appropriate fuel rod bow DNBR penalty (less than 1.5% per WCAP-8691, Revision 1). The rest of the margin between design and safety analysis DNBR limits can be used for plant design flexibility.

The RCS flow requirement is based on the loop minimum measured flow rate of 97,600 gpm which is used in the Improved Thermal Design Procedure described in FSAR 4.4.1 and 15.0.3. A precision heat balance is performed once each cycle and is used to calibrate the RCS flow rate indicators. Potential fouling of the feedwater venturi, which might not be detected, could bias the results from the precision heat balance in a non-conservative manner. Therefore, a penalty of 0.1% is assessed for potential feedwater venturi fouling. A maximum measurement uncertainty of 2.2% has been included in the loop minimum measured flow rate to account for potential undetected feedwater venturi fouling and the use of the RCS flow indicators for flow rate verification. Any fouling which might bias the RCS flow rate measurement greater than 0.1% can be detected by monitoring and trending various plant performance parameters. If detected, action shall be taken, before performing subsequent precision heat balance measurements, i.e., either the effect of fouling shall be quantified and compensated for in the RCS flow rate measurement, or the venturi shall be cleaned to eliminate the fouling.

Surveillance Requirement 4.2.3.4 provides adequate monitoring to detect possible flow reductions due to any rapid core crud buildup.

Surveillance Requirement 4.2.3.5 specifies that the measurement instrumentation shall be calibrated within seven days prior to the performance of the calorimetric flow measurement. This requirement is due to the fact that the drift effects of this instrumentation are not included in the flow measurement uncertainty analysis. This requirement does not apply for the instrumentation whose drift effects have been included in the uncertainty analysis.

HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

The limits of Section 3.2.3 for $F^N_{\Delta H}$ do not assume any specific uncertainty on the measured value of $F^N_{\Delta H}$. An appropriate uncertainty of 4% (nominal) or greater is added to the measured value of $F^N_{\Delta H}$ before it is compared with the requirement.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the Incore Detector Flux Mapping System, and a 3% allowance is appropriate for manufacturing tolerance.

The Radial Peaking Factor, $F_{xy}(Z)$ is measured periodically to provide assurance that the Hot Channel $F_Q(Z)$ remains within its limit. The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) as provided in Specification 3.2.2 was determined from expected power control maneuvers over the full range of burnup conditions in the core.

The 12-hour periodic surveillance of indicated RCS flow is sufficient to detect flow degradation which could lead to operation outside the acceptable limit.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 36 TO FACILITY OPERATING LICENSE NO. NPF-37

AND AMENDMENT NO. 36 TO FACILITY OPERATING LICENSE NO. NPF-66

COMMONWEALTH EDISON COMPANY

BYRON STATION, UNITS 1 AND 2

DOCKET NOS. 50-454 AND 50-455

TAC NOS. 74166 AND 74167

1.0 INTRODUCTION

By letter dated July 31, 1989 (Reference 1), Commonwealth Edison (the licensee) submitted a request for Technical Specification (TS) changes to allow refueling and operation of the Byron Station Unit 1 Cycle 4 and Unit 2 Cycle 3 cores with the VANTAGE 5 fuel design. Currently, both units of Byron Station are operating with a Westinghouse 17x17 optimized fuel assembly (OFA) core. Future core loadings consist of a mixed core of 50%-70% OFA and 30-50% VANTAGE 5 to eventually an all VANTAGE 5 fueled core. The VANTAGE 5 fuel design has been approved with conditions in the NRC Safety Evaluation on Westinghouse Topical Report WCAP-10444-P-A, "Reference Core Report VANTAGE 5 Fuel Assembly." The major design features of VANTAGE 5 fuel relative to the current OFA fuel design include: integral fuel burnable absorbers (IFBA), intermediate flow mixer grids (IFM), reconstitutable top nozzles, extended burnup capability, axial blankets and debris filter bottom nozzle. The licensee indicated in Reference 2 that the transition core and full VANTAGE 5 core safety analyses were performed at a thermal power level of 3411 MWt. Other assumptions included a full power F^N_H of 1.65 for the VANTAGE 5 fuel and 1.55 for the OFA fuel, an increase in the maximum F_Q to 2.50 and 10% steam generator tube plugging for the transient analysis and 15% for the LOCA analysis.

The TS changes include: (1) use of Westinghouse WRB-2 DNBR correlation for the VANTAGE 5 fuel, (2) an added maximum F^N_H of 1.65 for the VANTAGE 5 fuel, (3) an increased maximum F_Q of 2.50 from 2.32, and (4) an increased control rod drop time from 2.4 to 2.7 seconds.

During the review of the VANTAGE 5 fuel design in WCAP-10444-P-A, we identified conditions imposed on those licensees using the VANTAGE 5 fuel design. Our review of the licensee's request for the TS changes, the associated supporting analyses and the responses to the staff's review questions (Refs. 1 and 2) will address those conditions listed in the Safety Evaluation (SE) on WCAP-10444-P-A.

2.0 EVALUATION

2.1 Statistical Convolution Method

In the SE on WCAP-10444-P-A, we stated that the statistical method should not be used in VANTAGE 5 for evaluating the fuel rod shoulder gap. The licensee indicated (Ref. 2) that the statistical convolution method was not used for the VANTAGE 5 fuel design and the currently approved method was used for evaluating the fuel rod shoulder gap. Therefore, we consider this acceptable.

2.2 Seismic and LOCA Loads

In the SE on WCAP-10444-P-A, we stated that for each plant application, it must be demonstrated that the fuel assembly will maintain its coolable geometry under combined seismic and LOCA loads. The licensee performed LOCA and seismic load evaluations for transition cores and an all VANTAGE 5 core. The results indicate that the fuel assembly in either case has enough margin to sustain the combined seismic and LOCA loads such that the structural integrity and coolable geometry are maintained. Based on the licensee's evaluation results, we conclude that the condition of seismic and LOCA loads is satisfied.

2.3 Irradiation Demonstration Program

In the SE on WCAP-10444-P-A, we required that an irradiation program be performed to confirm the VANTAGE 5 fuel performance. The licensee indicated that there were numerous demonstration programs involving VANTAGE 5 fuel assemblies. During 1984 through 1988, four VANTAGE 5 demonstration assemblies were loaded into the V.C. Summer Unit 1 Cycle 2 and achieved an average burnup of about 46,000 MWD/MTU. Individual VANTAGE 5 product features have been demonstrated at other nuclear plants. IFBA demonstration fuel rods have been irradiated in Turkey Point Units 3 and 4 for two reactor cycles and the IFM grid feature has been irradiated at McGuire Unit 1 for three reactor cycles. The satisfactory performance of these demonstration assemblies resulted in the VANTAGE 5 fuel reload in many Westinghouse reactors. Thus, we conclude that VANTAGE 5 fuel will perform satisfactorily in the Byron Station.

2.4 Improved Thermal Design Procedure (ITDP)

In the SE on WCAP-10444-P-A, we stated that those restrictions in approving the use of the NRC approved Westinghouse improved thermal design procedures, ITDP (Ref. 3), should be applied to the VANTAGE 5 fuel design. The licensee indicated (Ref. 2) that they complied with the restrictions of ITDP for Byron. We therefore conclude that this is acceptable.

2.5 Positive Moderator Temperature Coefficient

In the SE on WCAP-10444-P-A, we stated that if a positive moderator temperature coefficient (MTC) is intended, the same MTC should be used in the plant-specific analysis. The licensee indicated (Ref. 2) that the MTC will not be positive for the analyzed cycles. Thus, we conclude that this restriction is satisfactorily met.

2.6 Transient Analysis

In the SE on WCAP-10444-P-A, we required that plant-specific analysis be performed to show that the appropriate safety criteria are not violated with the higher value of F_H^N and use of the VANTAGE 5 fuel. The licensee evaluated all the transient analyses for Byron Units 1 and 2 upgraded to VANTAGE 5 fuel and plant operation with an increased maximum F_H^N (from 1.55 to 1.65), an increased maximum F_Q (from 2.32 to 2.50) and an increased control rod drop time (from 2.4 to 2.7 seconds). The licensee also assumed the steam generator tube plugging to a level of 10% in their evaluation. The licensee determined the events affected significantly by the fuel design updates and operating condition changes and reanalyzed those events. In Reference 1, the licensee presented the reanalyzed results for the transients to support the reload application and technical specification changes.

The reanalyzed events can be summarized into three categories:

(1) DNBR transients affected by increase of F_H^N. The events are partial loss of flow, complete loss of flow, RCP shaft break and RCP locked rotor with loss of offsite power.

(2) The transients affected by increase of F_Q. The transients are RCP locked rotor and rod ejection.

(3) The transients affected by increase of the control rod drop time. The events are RCP locked rotor and rod ejection.

The licensee determined that for this application, the minimum required DNBR values for the OFA fuel analysis are 1.32 for thimble cold wall cells (three fuel rods and a thimble tube) and 1.34 for a typical cell (four fuel rods). The design DNBR values for the VANTAGE 5 fuel are 1.32 and 1.33 for thimble and typical cells, respectively. However, in order to demonstrate that the design DNBR values have enough margin to accommodate fuel rod bow penalty and effect of the mixed cores, the licensee determined that the minimum operating DNBR limits are 1.47 for thimble and 1.49 for typical cells for OFA fuel, and 1.65 and 1.67 for thimble and typical cells respectively for VANTAGE 5.

Since the licensee used the NRC approved methods to show that all applicable transient analysis acceptance criteria will not be violated for the proposed cycles, we approve the transient analyses.

2.7 Reactor Coolant Pump Shaft Seizure

In the SE on WCAP-10444-P-A, we stated that the mechanistic approach in determining the fraction of the fuel failures during the reactor pump seizure accident was unacceptable and the fuel failure criteria should be 95/95 DNBR limit. The licensee reanalyzed the reactor coolant pump shaft seizure (locked rotor) accident based on a failure criterion of the peak clad temperature of 2700°F. The licensee concluded that there is no fuel failure and the coolability was maintained since the calculated peak clad temperature (1853°F) remained much less than 2700°F and the amount of Zirconium-water reaction was small. As indicated above, we disapprove of the use of a mechanistic approach based on 2700°F peak clad temperature in determining the fuel failure. In response (Ref. 2), the licensee indicated that this event was analyzed by using the previously approved methods and showed that no rod was predicted to be below the 95/95 DNBR limit. Since the acceptable fuel failure criterion of 95/95 DNBR limit is used for DNBR analysis, we conclude that the reactor coolant pump shaft seizure accident is satisfactorily addressed for VANTAGE 5 fuel.

2.8 LOCA Analysis

In the SE on WCAP-10444-P-A, we stated that the plant specific analysis should be performed to show that the requirements of 10 CFR 50.46 are met. The licensee analyzed large and small break LOCAs to support the reload licensing application. In the licensee's large break LOCA analysis (Ref. 1), only double end cold leg guillotine (DECLG) breaks were analyzed since they were identified previously as limiting cases that result in the highest peak clad temperature. The DECLG break analysis was performed with a total peaking factor of 2.5, 102% of the core power of 3411 MWt, temperatures between 600 to 619.3°F in the RCS hot legs and 535.6 to 556.7 in the RCS cold legs respectively, and an assumed loss of offsite power at the beginning of the accident. An assumption of 15% steam generator tube plugging was made for the analysis. A sensitivity study of DECLG break sizes on the effect of the peak clad temperature was performed by use of discharge coefficients of 0.8, 0.6, and 0.4. The results showed that the DECLG break with a discharge coefficient of 0.6 with the RCS operating at a nominal hot leg temperature of 619.3°F is the worst large break case resulting in a peak clad temperature of 1883.1°F. Analysis performed assuming the RCS to be operating with a reduced hot leg temperature of 600°F was found to be less limiting than the result obtained when the RCS was assumed to be operating with a hot leg temperature of 619.3°F. The licensee evaluated the effect of transition core cycles on the calculated PCT and determined that the maximum increase in PCT is 50°F which yields a transition core PCT of 1933.1°F.

The analysis of a large break LOCA transient is divided into three phases: (1) blowdown, (2) refill, and (3) reflood. The licensee used SATAN-V1 code (Ref. 4) for the transient thermal hydraulic calculation during blowdown period; the WREFLOOD (Ref. 5) and BASH codes (Ref. 6) for the thermal hydraulic calculation of refill and reflood transient periods; the LOCBART code (Ref. 7) for calculation of peak clad temperature and the COCO code (Ref. 8) for the calculation of containment pressure transient.

As a result of our review, we find that the approved analytical models and computer codes were used and results showed that the peak clad temperature of 1933.1°F, total metal-water reaction of less than 0.3% of the fuel clad and local clad oxidation of less than 3.26% are within the 10 CFR 50.46 acceptable criteria which are 2200°F, 1% and 17%, respectively. Therefore, we conclude that the large break LOCA analysis is acceptable.

In the licensee's small break LOCA analysis, we find that the licensee used the approved NOTRUMP code (Refs. 9 and 10) for the calculation of transient depressurization of the reactor coolant system and core power and the LOCTA code (Refs. 7) for the calculation of the peak clad temperature. Only one core flow channel is modeled in NOTRUMP since the core flow during a small break is relatively slow, providing enough time to maintain flow equilibrium between fuel assemblies (i.e., no crossflow) in mixed cores. Hydraulic resistance mismatch is not a factor for small break. Therefore, the licensee referenced the small break LOCA for the complete core of the VANTAGE 5 fuel design as the bounding case for all transition cycles. The analysis was done with assumptions of 102% of the core power of 3411 MWt and a total peaking factor of 2.5. Analyses for these break sizes were performed to show that the worst break size is a 3-inch diameter break which results in the highest peak clad temperature of 1453.1°F, well below the acceptable criterion of 2200°F. Since the approved methods were used to show the analytical results to be within the acceptance criteria imposed in 10 CFR 50.46, we therefore conclude that the small break LOCA analysis is acceptable.

3.0 TECHNICAL SPECIFICATION CHANGES

The proposed technical specification (TS) changes reflect impact of the fuel design change and assumptions used in the safety analysis to support the reload application. We discuss the TS changes as follows.

(1) New DNBR Correlation and Operating DNBR Limits - pp B 2-1, B 3/4 2-1, B 3/4 2-4.

A new DNBR correlation of WRB-2 is added and the cycle specific operating DNBR limits with inclusion of the rod bow penalty factor and effect of the mixed core are added to the TS. Since the changes are consistent with the assumptions used in the transient analysis, we approve the changes.

(2) Increased Control Rod Drop Time - p 3/4 1-19.

The control rod drop time is revised to 2.7 seconds from 2.4 seconds due to the use of the VANTAGE 5 fuel design. The licensee has taken into account the effect of the increased control rod drop time in all related safety analysis. Thus, we conclude that this change is acceptable.

(3) Increased Peaking Factor - F_H^N (pp B 2-2, 3/4 2-8, B 3/4 2-5)
F_Q (pp 3/4 2-4, B 3/4 2-1)

The maximum F_H^N and F_Q are increased from 1.55 to 1.65, and 2.32 to 2.50, respectively. Since the changes are consistent with the assumptions used in the analyses to support the reload application, we approve the changes.

(4) VANTAGE 5 Design - pp 2-8, 3/4 2-7.

The VANTAGE 5 fuel design is added to the TS. Since VANTAGE 5 is acceptable for use in the Byron cores, we conclude that the changes are acceptable.

(5) Surveillance Requirement Changes - (pp 3/4 1-4, 3/4 1-5, B 3/4 1-2).

BOL is deleted from MTC LCO. Surveillance 4.1.1.3.a is modified to compare BOL MTC with burnup and rod withdrawal limits are developed to keep MTC negative. Since the changes are supported by the analytical assumption that no positive MTC is used through the cycle, we approve the changes.

We have reviewed the licensee submittal of technical specification changes and related analytical results to support the request to allow the operation of Cycles 4 and 3 of Byron Units 1 and 2 cores, respectively. Based on the approved generic topical report, WCAP-10444-P-A, and plant specific analyses (Ref. 1), we approve the use of VANTAGE 5 fuel design and technical specification changes for the Byron Station Unit 1 Cycle 3 and Unit 2 Cycle 4 reload cores.

4.0 FINDING OF NO SIGNIFICANT IMPACT

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact has been prepared and published (55 FR 3123) in the Federal Register on January 30, 1990. Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

5.0 CONCLUSION

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

6.0 REFERENCES

1. Letter from R. Chrzanowski (Commonwealth Edison) to T. Murley (NRC), dated July 31, 1989.

2. Letter from R. Chrzanowski (Commonwealth Edison) to T. Murley (NRC), dated October 19, 1989.

3. WCAP-8567-P-A, Improved Thermal Design Procedure, February 1989.

4. WCAP-8302 (Proprietary) and WCAP-8306 (Non-Proprietary), SATAN-VI Program: Comprehensive Spacetime Dependent Analysis of Loss of Coolant, June 1974.

5. WCAP-8170 (Proprietary) and WCAP-8171 (Non-Proprietary), Calculated Model for Code Reflood After a Loss of Coolant Accident (WREFLOOD), June 1974.

6. WCAP-10266-P-A, Revision 2 with Addenda (Proprietary), the 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code, August 1986.

7. WCAP-8301 (Proprietary) and WCAP-8305 (Non-Proprietary), LOCTA-IV Program: Loss of Coolant Transient Analysis, June 1974.

8. WCAP-8327 (Proprietary) and WCAP-8326 (Non-Proprietary), Containment Pressure Analysis Code (COCO), June 1974.

9. WCAP-10079-P-A (Proprietary) and WCAP-10080-P-A (Non-Proprietary), NOTRUMP, A Nodal Transient Small Break and General Network Code, August 1985.

10. WCAP-10054-P-A (Proprietary) and WCAP-10081-P-A (Non-Proprietary), Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code, August 1985.

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