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

October 20, 1989

Docket Nos. 50-254 and 50-265

Mr. Thomas J. Kovach
Nuclear Licensing Manager
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Kovach:

SUBJECT: TECHNICAL SPECIFICATION AMENDMENT AND CORE OPERATING LIMITS REPORT PER GENERIC LETTER 88-16 (TAC NOS. 73761 AND 73762)

The Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 120 and 116 to Appendix A, Technical Specifications, of Facility Operating License Nos. DPR-29 and DPR-30 for the Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2. These amendments are in response to an application from Commonwealth Edison Company dated July 11, 1989, as supplemented on August 14, 1989.

These amendments revise QCNPS Technical Specifications, in accordance with Generic Letter 88-16, by replacing cycle-specific operating limits with references to a Core Operating Limits Report.

Our related Safety Evaluation report is also enclosed. A Notice of Issuance for these amendments will be included in the Commission's biweekly Federal Register notices.

Sincerely,

Thierry M. Ross

Thierry M. Ross, Project Manager
Project Directorate III-2
Division of Reactor Projects - III, IV, V, and Special Projects
Office of Nuclear Reactor Regulation

CP-1

Enclosures:
1. Amendment No. 120 to License No. DPR-29
2. Amendment No. 116 to License No. DPR-30
3. Safety Evaluation

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cc w/enclosures:
See next page

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October 20, 1989

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9/25/89	9/28/89	9/03/89	9/28/89

QUAD AMEND 73761/2

Mr. Thomas J. Kovach
Commonwealth Edison Company

Quad Cities Nuclear Power Station
Units 1 and 2

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

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-254

QUAD CITIES NUCLEAR POWER STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 120
License No. DPR-29

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated July 11, 1989, as supplemented by August 14, 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 that (i) activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Facility Operating License No. DPR-29 is hereby amended to read as follows:

8911020166

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 120, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Paul C. Shemanski

Paul C. Shemanski, Acting Director
Project Directorate III-2
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: October 20, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 120

FACILITY OPERATING LICENSE NO. DPR-29

DOCKET NO. 50-254

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
vii	vii
1.0-1	1.0-1
1.1/2.1-7	1.1/2.1-7
1.1/2.1-10	1.1/2.1-10
3.2/4.2-19	3.2/4.2-19
3.2/4.2-20	3.2/4.2-20
3.3/4.3-7	3.3/4.3-7
3.3/4.3-14	3.3/4.3-14
3.5/4.5-13	3.5/4.5-13
3.5/4.5-14	3.5/4.5-14
3.5/4.5-20	3.5/4.5-20
3.5/4.5-26	3.5/4.5-26
Figure 3.5-1 (sheets 1 thru 5)	
Figure 3.5-2	
3.6/4.6-11	3.6/4.6-11
3.6/4.6-24	3.6/4.6-24
6.6-2	6.6-2
6.6-3	6.6-3
6.6-4	6.6-4

QUAD CITIES
DPR-29

TECHNICAL SPECIFICATIONS

APPENDIX A

LIST OF FIGURES

Number	Title
2.1-1	APRM Flow Reference Scram and APRM Rod Block Settings
2.1-2	Deleted
2.1-3	APRM Flow Bias Scram Relationship to Normal Operating Conditions
4.1-1	Graphical Aid in the Selection of an Adequate Interval Between Tests
4.2-1	Test Interval vs. System Unavailability
3.4-1	Deleted
3.4-2	Sodium Pentaborate Solution Temperature Requirements
3.5-1	Deleted
3.5-2	Deleted
3.6-1	Minimum Temperature Requirements per Appendix G of 10 CFR 50
3.6-2	Minimum Reactor Pressurization Temperature
4.6-1	Chloride Stress Corrosion Test Results at 500°F
4.8-1	Locations of Fixed Environmental Radiological Monitoring Stress
6.1-1	Deleted
6.1-2	Deleted
6.1-3	Minimum Shift Manning Chart

1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

A. Alteration of the Reactor Core - The act of moving any component in the region above the core support plate, below the upper grid, and within the shroud. Normal control rod movement with the control rod drive hydraulic system is not defined as a core alteration. Normal movement of incore instrumentation or movement of the TIP system is not defined as a core alteration.

B. CORE OPERATING LIMITS REPORT - The unit specific document that provides core operating limits for the current operating reload cycle. These cycle specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.6. Plant operation within these operating limits is addressed in individual specifications.

C. Hot Standby - Hot standby means operation with the reactor critical, system pressure less than 1060 psig, the main steam isolation valves closed, and thermal power not exceeding 15%.

D. Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.

E. Instrument Calibration - An instrument calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range and accuracy, to a known value (values) of the parameter which the instrument monitors. Calibration shall encompass the entire instrument, including actuation, alarm, or trip. Response time is not part of the routine instrument calibration but will be checked once per operating cycle.

F. Instrument Functional Test - An instrument functional test means the injection of a simulated signal into the instrument primary sensor to verify the proper instrument response alarm and/or initiating action.

G. Instrument Check - An instrument check is qualitative determination of acceptable operability by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.

The MCPR fuel cladding integrity safety limit has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from the normal operation condition, more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the safety limit, is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state, including uncertainty in the boiling transition correlation (see e.g., Reference 1). Because the boiling transition correlation is based on a large quantity of full-scale data, there is a very high confidence that operation of a fuel assembly at the condition of MCPR = the fuel cladding integrity safety limit would not produce boiling transition.

However, if boiling transition were to occur, cladding perforation would not be expected. Cladding temperature would increase to approximately 1100°F, which is below the perforation temperature of the cladding material. This had been verified by tests in the General Electric Test Reactor (GETR), where similar fuel operated above the critical heat flux for a significant period of time (30 minutes) without cladding perforation.

If reactor pressure should ever exceed 1400 psia during normal power operation (the limit of applicability of the boiling transition correlation), it would be assumed that the fuel cladding integrity safety limit has been violated.

In addition to the boiling transition limit (MCPR) operation is constrained to a maximum LHGR specified in the CORE OPERATING LIMITS REPORT for various fuel types. This constraint is established by Specification 3.5.J. to provide adequate safety margin to 1% plastic strain for abnormal operating transients initiated from high power conditions. Specification 2.1.A.1 provides for equivalent safety margin for transients initiated from lower power conditions by adjusting the APRM flow-biased scram setting by the ratio of FRP/MFLPD.

Specification 3.5.J established the LHGR maximum which cannot be exceeded under steady power operation.

B. Core Thermal Power Limit (Reactor Pressure < 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

2.1 LIMITING SAFETY SYSTEM SETTING BASES

The abnormal operational transients applicable to operation of the units have been analyzed throughout the spectrum of planned operating conditions in accordance with Regulatory Guide 1.49. In addition, 2511 MWt is the licensed maximum steady-state power level of the units. This maximum steady-state power level will never knowingly be exceeded.

Conservatism incorporated into the transient analysis is documented in References 1 and 2. Transient analyses are initiated at the conditions given in these References.

The scram delay time and rate of rod insertion allowed by the analyses are conservatively set equal to the longest delay and slowest insertion rate acceptable by technical specifications. The effects of scram worth, scram delay time, and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity is assured by the time requirements for 5% and 20% insertion. By the time the rods are 60% inserted, approximately 4 dollars of negative reactivity have been inserted, which strongly turns the transient and accomplishes the desired effect. The times for 50% and 90% insertion are given to assure proper completion of the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

The MCPR operating limit is, however, adjusted to account for the statistical variation of measured scram times as discussed in Reference 2 and the bases of Specification 3.5.K.

Steady-state operation without forced recirculation will not be permitted except during startup testing. The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculation pumps.

The bases for individual trip settings are discussed in the following paragraphs.

For analyses of the thermal consequences of the transients, the MCPR's stated in the CORE OPERATING LIMITS REPORT as the limiting condition of operation bound those which are conservatively assumed to exist prior to initiation of the transients.

TABLE 3.2-3

INSTRUMENTATION THAT INITIATES ROD BLOCK

Minimum Number of Operable or Tripped Instrument Channels per Trip System [1]	Instrument	Trip Level Setting
2	APRM upscale (flow bias)[7]	$\leq(0.58W_D + 50)\frac{FRP}{MFLPD}$ [2]
2	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]	[10]
1	Rod block monitor downscale[7]	$\geq$3/125 full scale
3	IRM downscale[3] [8]	$\geq$3/125 full scale
3	IRM upscale[8]	$\leq$108/125 full scale
2[5]	SRM detector not in Startup position [4]	$\geq$2 feet below core centerline
3	IRM detector not in Startup position [8]	$\geq$2 feet below core centerline
2[5] [6]	SRM upscale	$\leq 10^5$ counts/sec
2[5]	SRM downscale [9]	$\geq 10^2$ counts/sec
1 (per bank)	High water level in scram discharge volume (SDV)	$\leq$ 25 gallons (per bank)
1	SDV high water level scram trip bypassed	NA

TABLE 3.2-3 (Con't)

Notes

[1] For the Startup/Hot Standby and Run positions of the reactor mode selector switch, there shall be two operable or tripped trip systems for each function except the SRM rod blocks. IRM upscale and IRM downscale need not be operable in the Run position, APRM downscale, APRM upscale (flow biased), and RBM downscale need not be operable in the Startup/Hot Standby mode. The RBM upscale need not be operable at less than 30% rated thermal power. One channel may be bypassed above 30% rated thermal power provided that a limiting control rod pattern does not exist. For systems with more than one channel per trip system, if the first column cannot be met for one of the two trip systems, this condition may exist for up to 7 days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than 7 days the system shall be tripped. If the first column cannot be met for both trip systems, the systems shall be tripped.

[2] W_D is the percent of drive flow required to produce a rated core flow of 98 million lb/hr. Trip level setting is in percent of rated power (2511 MWt).

[3] IRM downscale may be bypassed when it is on its lowest range.

[4] This function is bypassed when the count rate is $\geq$ 100CPS.

[5] One of the four SRM inputs may be bypassed.

[6] This SRM function may be bypassed in the higher IRM ranges (ranges 8, 9, and 10) when the IRM upscale rod block is operable.

[7] Not required to be operable while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5MWt.

[8] This IRM function occurs when the reactor mode switch is in the Refuel or Startup/Hot Standby position.

[9] This trip is bypassed when the SRM is fully inserted.

[10] The Rod Block Monitor upscale setpoint shall be established as specified in the CORE OPERATING LIMITS REPORT.

interval not more frequently than 16 weeks nor less frequently than 32 weeks. These tests shall be performed with a reactor pressure above 800 psig and may be measured during a reactor scram. Whenever all of the control rod drive scram times have been measured, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained. The results of measurements performed on the control rod drives shall be submitted in the annual operating report to the NRC.

3. If Specification 3.3.C.1 cannot be met, the reactor shall not be made supercritical; if operating, the reactor shall be shut down immediately upon determination that average scram time is deficient.

4. If Specification 3.3.C.2 cannot be met, the deficient control rod shall be considered inoperable, fully inserted into the core, and electrically disarmed.

5. If the overall average of the 20% insertion scram time data generated to date in the current cycle exceeds the limit specified in the CORE OPERATING LIMITS REPORT, the MCPR operating limit must be modified as required by Specification 3.5.K.

3. The cycle cumulative mean scram time for 20% insertion will be determined immediately following the testing required in Specifications 4.3.C.1 and 4.3.C.2 and the MCPR operating limit adjusted, if necessary, as required by Specification 3.5.K.

D. Control Rod Accumulators

At all reactor operating pressures, a rod accumulator may be inoperable provided that no other control rod in the nine-rod square array around that rod has:

1. an inoperable accumulator,

D. Control Rod Accumulators

Once a shift, check the status of the pressure and level alarms for each accumulator.

6. The operability of the Scram Discharge Volume vent and drain valves assures the proper venting and draining of the Volume, so that water accumulation in the Volume does not occur. These specifications provide for the periodic verification that the valves are open, and for the testing of these valves under reactor scram conditions during each Refueling Outage.

C. Scram Insertion Times

The control rod system is analyzed to bring the reactor subcritical at a rate fast enough to prevent fuel damage, i.e., to prevent the MCPR from becoming less than the fuel cladding integrity safety limit.

Analysis of the limiting power transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above specification, provide the required protection, and MCPR remains greater than the fuel cladding integrity safety limit. It is necessary to raise the MCPR operating limit (per Specification 3.5.K) when the average 20% scram insertion time reaches the limit specified in the CORE OPERATING LIMITS REPORT on a cycle cumulative basis (overall average of surveillance data to date) in order to comply with assumptions in the implementation procedure for the ODYN transient analysis computer code. The basis for choosing this 20% scram insertion time limit is discussed further in the bases for Specification 3.5.K. In the analytical treatment of the transients, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typically observed time delay of about 210 milliseconds. Approximately 90 milliseconds after neutron flux reaches the trip point, the pilot scram valve solenoid deenergizes and 120 milliseconds later the control rod motion is estimated to actually begin. However, 200 milliseconds rather than 120 milliseconds is conservatively assumed for this time interval in the transient analyses and is also included in the allowable scram insertion times specified in Specification 3.3.C.

The scram times for all control rods will be determined at the time of each refueling outage. A representative sample of control rods will be scram tested during the interval of greater than 16 weeks but not more than 32 weeks.

Scram times of new drives are approximately 2.5 to 3 seconds; lower rates of change in scram times following initial plant operation at power are expected. The test schedule provides reasonable assurance of detection of slow drives before system deterioration beyond the limits of Specification 3.3.C. The program was developed on the basis of the statistical approach outlined below and judgment.

The history of drive performance accumulated to date indicates that the 90% insertion times of new and overhauled drives approximate a normal distribution about the mean which tends to become skewed toward longer

I. Average Planar LHGR

During steady-state power operation, the average linear heat generation rate (APLHGR) of all the rods in any fuel assembly, as a function of average planar exposure, at any axial location, shall not exceed the maximum average planar LHGR specified in the CORE OPERATING LIMITS REPORT. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned to within the prescribed limits within 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.

I. Average Planar LHGR

Daily during steady-state operation above 25% rated thermal power, the average planar LHGR shall be determined.

J. Local LHGR

During steady-state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR specified in the CORE OPERATING LIMITS REPORT. If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within 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.

J. Local LHGR

Daily during steady-state power operation above 25% of rated thermal power, the local LHGR shall be determined.

K. Minimum Critical Power Ratio (MCPR)

During steady-state operation at rated core flow, MCPR shall be equal to or greater than the MCPR limit specified in the CORE OPERATING LIMITS REPORT.

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 specified in the CORE OPERATING LIMITS REPORT. If any time during operation it is determined by normal surveillance 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 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.

K. Minimum Critical Power Ratio (MCPR)

The MCPR shall be determined daily during steady-state power operation above 25% of rated thermal power.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat-generation rate of all the rods of a fuel assembly at any axial location and is only secondarily dependent on the rod-to-rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak cladding temperature by less than ±20°F relative to the peak temperature for a typical fuel design, the limit on the average planar LHGR is sufficient to assure that calculated temperatures are below the limit. The maximum average planar LHGR's specified in the CORE OPERATING LIMITS REPORT are based on calculations employing the models described in Reference 2.

The Average Planar Linear Heat Generation Rate (APLHGR) also serves a secondary function which is to assure fuel rod mechanical integrity.

J. Local LHGR

This specification assures that the maximum linear heat-generation rate in any rod is less than the design linear heat-generation rate specified in the CORE OPERATING LIMITS REPORT even if fuel pellet densification is postulated. The power spike penalty is discussed in Reference 2 and assumes a linearly increasing variation in axial gaps between core bottom and top and assures with 95% confidence that no more than one fuel rod exceeds the design LHGR due to power spiking. No penalty is required in Specification 3.5.L because it has been accounted for in the reload transient analyses by increasing the calculated peak LHGR by 2.2%.

K. Minimum Critical Power Ratio (MCPR)

The steady state values for MCPR specified in the CORE OPERATING LIMITS REPORT 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 plus two percent for uncertainty is satisfied. For any of the special set of transients or disturbances 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 value of MCPR stated in the CORE OPERATING LIMITS REPORT for the limiting condition of operation bounds the initial value of MCPR assumed to exist prior to the initiation of the transients. This initial condition, which is used in the transient analyses, will preclude violation of the fuel cladding integrity safety limit. Assumptions and methods used in calculating the required steady state MCPR limit for each reload cycle are documented in References 2 and 4. The results apply with increased conservatism while operating with MCPRs greater than specified.

Operation of the fans and coolers is required during shutdown and thus additional surveillance is not required.

Verification that access doors to each vault are closed following entrance by personnel is covered by station operating procedures.

The LHGR shall be checked daily to determine if fuel burnup or control rod movement has caused changes in power distribution. Since changes due to burnup are slow and only a few control rods are moved daily, a daily check of power distribution is adequate.

Average Planar LHGR

At core thermal power levels less than or equal to 25%, operating plant experience and thermal hydraulic analyses indicate that the resulting average planar LHGR is below the maximum average planar LHGR by a considerable margin; therefore, evaluation of the average planar LHGR below this power level is not necessary. The daily requirement for calculating average planar LHGR above 25% rated thermal power is sufficient, since power distribution shifts are slow when there have not been significant power or control rod changes.

Local LHGR

The LHGR as a function of core height shall be checked daily during reactor operation at greater than or equal to 25% power to determine if fuel burnup or control rod movement has caused changes in power distribution. A limiting LHGR value is precluded by considerable margin when employing any permissible control rod pattern below 25% rated thermal power.

Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicate that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR.

The daily requirement for calculating MCPR above 25% rated thermal power is sufficient, since power distribution shifts are very slow when there have not been significant power or control rod changes. In addition, the K_f correction, as specified in the CORE OPERATING LIMITS REPORT, applied to the LCO provides margin for flow increases from low flows.

2. If Specification 3.6.H.1 cannot be met, one recirculation pump shall be tripped.

3. During Single Loop Operation for more than 12 hours, the following restrictions are required:

   a. The MCPR Safety Limit shall be increased by 0.01 (T.S. 1.1A);

   b. The MCPR Operating Limit, as specified in the CORE OPERATING LIMITS REPORT, shall be increased by 0.01 (T.S. 3.5.K);

   c. The flow biased APRM Scram and Rod Block Setpoints shall be reduced by 3.5% to read as follows:

      T.S. 2.1.A.1;
      $S \leq .58WD + 58.5$

      T.S. 2.1.A.1;*
      $S \leq (.58WD + 58.5)$ FRP/MFLPD

      T.S. 2.1.B;
      $S \leq .58WD + 46.5$

      T.S. 2.1.B;*
      $S \leq (.58WD + 46.5)$ FRP/MFLPD

      T.S. 3.2.C (Table 3.2-3);*
      APRM Upscale $\leq (.58WD + 46.5)$ FRP/MFLPD

      * In the event that MFLPD exceeds FRP.

   d. The flow biased RBM Rod Block setpoints, as specified in the CORE OPERATING LIMITS REPORT, shall be reduced by 4.0%.

   e. The suction valve in the idle loop shall be closed and electrically isolated except when the idle loop is being prepared for return to service.

H. Recirculation Pump Flow Limitations

The LPCI loop selection logic is described in the SAR, Section 6.2.4.2.5. For some limited low probability accidents with the recirculation loop operating with large speed differences, it is possible for the logic to select the wrong loop for injection. For these limited conditions, the core spray itself is adequate to prevent fuel temperatures from exceeding allowable limits. However, to limit the probability even further, a procedural limitation has been placed on the allowable variation in speed between the recirculation pumps.

The licensee's analyses indicate that above 80% power the loop select logic could not be expected to function at a speed differential of 15%. Below 80% power, the loop select logic would not be expected to function at a speed differential of 20%. This specification provides a margin of 5% in pump speed differential before a problem could arise. If the reactor is operating on one pump, the loop select logic trips that pump before making the loop selection.

Analyses have been performed which support indefinite single loop operation provided the appropriate restrictions are implemented within 12 hours. The MCPR Safety Limit has been increased by 0.01 to account for core flow and TIP reading uncertainties which are used in the statistical analysis of the safety limit. The MCPR Operating Limit, as specified in the CORE OPERATING LIMITS REPORT, has also been increased by 0.01 to maintain the same margin to the safety limit as during Dual Loop operation.

The flow biased scram and rod block setpoints are reduced to account for uncertainties associated with backflow through the idle jet pumps when the operating recirculation pump is above 20-40% of rated speed. This assures that the flow biased trips and blocks occur at conservative neutron flux levels for a given core flow.

The closure of the suction valve in the idle loop prevents the loss of LPCI flow through the idle recirculation pump into the downcomer.

I. Snubbers

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on non-safety-related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety-related system.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

whole body dose received from external sources shall be assigned to specific major work functions.

3. Monthly Operating Report

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office of Management Information and Program Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555, with a copy to the appropriate Regional Office, to arrive no later than the 15th of each month following the calendar month covered by the report. In addition, any changes to the ODCM shall be submitted with the Monthly Operating Report within 90 days of the effective date of the change.

A report of major change to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted by the onsite review function. If such change is re-evaluated and not installed, notification of cancellation of the change should be provided to the NRC.

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 Rod Withdrawal Block Monitor Upscale Instrumentation Setpoint for Table 3.2-3 of Specification 3.2.C and for Specification 3.6.H.

(2) The overall average of the 20% insertion scram time data for Specification 3.3.C.

(3) The Average Planar Heat Generation Rate (APLHGR) for Specification 3.5.I.

(4) The Linear Heat Generation Rate (LHGR) for Specification 3.5.J.

(5) The Minimum Critical Power Ratio (MCPR) for Specification 3.5.K and 3.6.H.

(6) The K_f core flow MCPR adjustment factor for Specification 3.5.K.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel (latest approved revision).

c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

d. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

B. Unique Reporting Requirements

1. Radioactive Effluent Release Report (Semi-Annual)

A semi-annual report shall be submitted to the Commission within 60 days after January 1 and July 1 of each year specifying the quantity of each of the radionuclides released to unrestricted areas in liquid and gaseous effluents during the previous 6 months. The format and content of the report shall be in accordance with Regulatory Guide 1.21 (Revision 1) dated June, 1974. Any changes to the PCP shall be included in this report.

2. Environmental Program Data (Annual Report)

An annual report containing the data taken in the standard radiological monitoring program (Table 4.8-4) shall be submitted prior to May 1 of each year. The content of the report shall include:

a. Results of all environmental measurements summarized in the format of the Regulatory Guide 4.8 Table 1 (December 1975). (Individual sample results will be retained at the Station). In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. Summaries, interpretations, and analysis of trends of the results are to be provided.

b. An assessment of the monitoring results and radiation dose via the principal pathways of exposure resulting from plant emissions of radioactivity including the maximum noble gas gamma and beta air doses in the unrestricted area. The assessment of radiation doses shall be performed in accordance with the Offsite Dose Calculation Manual (ODCM).

c. Results of the census to determine the locations of nearest residences and of nearby animals producing milk for human consumption (Table 4.8-4).

d. The reason for the omission if the nearest dairy to the station is not in the monitoring program (Table 4.8-4).

e. An annual summary of meteorological conditions concurrent with the releases of gaseous effluents in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.

f. The results of the Interlaboratory Comparison Program described in section 3.8.D.7.

g. The results of the 40 CFR 190 uranium fuel cycle dose analysis for each calendar year.

h. A summary of the monitoring program, including maps showing sampling locations and tables giving distance and direction of sampling locations from the Station.

3. If a confirmed measured radionuclide concentration in an environmental sampling medium averaged over any calendar quarter sampling period exceeds the reporting level given in Table 4.8-5 and if the radioactivity is attributable to plant operation, a written report shall be submitted to the Administrator of the NRC Regional Office, with a copy to the Director, Office of Nuclear Reactor Regulation, within 30 days from the end of the quarter.

a. When more than one of the radionuclides in Table 4.8-5 are detected in the medium, the reporting level shall have been exceeded if

$$\Sigma \frac{Ci}{R.L.i} \geq 1$$

where Ci is the average quarterly concentration of the i^{th} radionuclide in the medium and RL is the reporting level of radionuclide i.

b. If radionuclides other than those in Table 4.8-5 are detected and are due to plant effluents, a reporting level is exceeded if the potential annual dose to an individual is equal to or greater than the design objective doses of 10 CFR 50, Appendix I.

c. This report shall include an evaluation of any release conditions, environmental factors, or other aspects necessary to explain the anomalous effect.

4. Special Reports

Special Reports shall be submitted as indicated in Table 6.6-1.

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

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-265

QUAD CITIES NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 116
License No. DPR-30

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated July 11, 1989, as supplemented by August 14, 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 that (i) activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) such activities will be conducted in compliance with the Commission's regulations;

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Facility Operating License No. DPR-30 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 116, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Paul C. Shemanski

Paul C. Shemanski, Acting Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 20, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 116

FACILITY OPERATING LICENSE NO. DPR-30

DOCKET NO. 50-265

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
vi	vi
1.0-1	1.0-1
1.1/2.1-4	1.1/2.1-4
1.1/2.1-7	1.1/2.1-7
3.2/4.2-14	3.2/4.2-14
3.2/4.2-14a	3.2/4.2-14a
3.3/4.3-5	3.3/4.3-5
3.3/4.3-10	3.3/4.3-10
3.5/4.5-9	3.5/4.5-9
3.3/4.5-9a	
3.5/4.5-10	3.5/4.5-10
3.5/4.5-13	3.5/4.5-13
3.5/4.5-14	3.5/4.5-14
3.5/4.5-14a	3.5/4.5-14a
3.5/4.5-18	3.5/4.5-18
Figure 3.5-1 (sheets 1 thru 6)	
Figure 3.5-2	
3.6/4.6-5a	3.6/4.6-5a
3.6/4.6-13a	3.6/4.6-13a
6.6-2	6.6-2
	6.6-2a

TECHNICAL SPECIFICATIONS

APPENDIX A

LIST OF FIGURES

Number	Title
2.1-1	APRM Flow Reference Scram and APRM Rod Block Settings
2.1-2	Deleted
2.1-3	APRM Flow Bias Scram Relationship to Normal Operating Conditions
4.1-1	Graphical Aid in the Selection of and Adequate Interval Between Tests
4.2-1	Test Interval vs. System Unavailability
3.4-1	Deleted
3.4-2	Sodium Pentaborate Solution Temperature Requirements
3.5-1	Deleted
3.5-2	Deleted
3.6-1	Minimum Temperature Requirements per Appendix G of 10 CFR 50
4.6-1	Chloride Stress Corrosion Test Results at 500°F
4.8-1	Locations of Fixed Environmental Radiological Monitoring Stress
6.1-1	Deleted
6.1-2	Deleted
6.1-3	Minimum Shift Manning Chart

1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

A. Alteration of the Reactor Core - The act of moving any component in the region above the core support plate, below the upper grid, and within the shroud. Normal control rod movement with the control rod drive hydraulic system is not defined as a core alteration. Normal movement of incore instrumentation or movement of the TIP system is not defined as a core alteration.

B. CORE OPERATING LIMITS REPORT - The unit specific document that provides core operating limits for the current operating reload cycle. These cycle specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.6. Plant operations within these operating limits is addressed in individual specifications.

C. Hot Standby - Hot standby means operation with the reactor critical, system pressure less than 1060 psig, the main steam isolation valves closed, and thermal power not exceeding 15%.

D. Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.

E. Instrument Calibration - An instrument calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range and accuracy, to a known value (values) of the parameter which the instrument monitors. Calibration shall encompass the entire instrument, including actuation, alarm, or trip. Response time is not part of the routine instrument calibration but will be checked once per operating cycle.

F. Instrument Functional Test - An instrument functional test means the injection of a simulated signal into the instrument primary sensor to verify the proper instrument response alarm and/or initiating action.

G. Instrument Check - An instrument check is qualitative determination of acceptable operability by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.

1.1 SAFETY LIMIT BASIS

The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, a step-back approach is used to establish a safety limit such that the minimum critical power ratio (MCPR) is no less than the fuel cladding integrity safety limit MCPR > the fuel cladding integrity safety limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The fuel cladding is one of the physical boundaries which separate radioactive materials from the environs. The integrity of the fuel cladding is related to its relative freedom from perforations or cracking.

Although some corrosion or use-related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system safety settings. While fission product migration from cladding perforation is just as measurable as that from use-related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding safety limit is defined with margin to the conditions which would produce onset of transition boiling (MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation. Therefore, the fuel cladding integrity safety limit is established such that no calculated fuel damage shall result from an abnormal operational transient. Basis of the values derived for this safety limit for each fuel type is documented in References 1 and 2.

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

Onset of transition boiling results in a decrease in heat transfer from the cladding and therefore elevated cladding temperature and the possibility of cladding failure. However, the existence of critical power, or boiling transition is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR), which is the ratio for the bundle power which would produce 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 has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from the normal operation condition, more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the safety limit, is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state, including uncertainty in the boiling transition correlation (see e.g., Reference 1). Because the boiling transition correlation is based on a large quantity of full-scale data, there is a very high confidence that operation of a fuel assembly at the condition of MCPR - the fuel cladding integrity safety limit would not produce boiling transition.

However, if boiling transition were to occur, cladding perforation would not be expected. Cladding temperature would increase to approximately 1100°F, which is below the perforation temperature of the cladding material. This had been verified by tests in the General Electric Test Reactor (GETR), where similar fuel operated above the critical heat flux for a significant period of time (30 minutes) without cladding perforation.

If reactor pressure should ever exceed 1400 psia during normal power operation (the limit of applicability of the boiling transition correlation), it would be assumed that the fuel cladding integrity safety limit has been violated.

In addition to the boiling transition limit (MCPR) operation is constrained to a maximum LHGR specified in the CORE OPERATING LIMITS REPORT for various fuel types. This constraint is established by Specification 3.5.J. to provide adequate safety margin to 1% plastic strain for abnormal operating transients initiated from high power conditions. Specification 2.1.A.1 provides for equivalent safety margin for transients initiated from lower power conditions by adjusting the APRM flow-biased scram setting by the ratio of FRP/MFLPD.

2.1 LIMITING SAFETY SYSTEM SETTING BASES

The abnormal operational transients applicable to operation of the units have been analyzed throughout the spectrum of planned operating conditions in accordance with Regulatory Guide 1.49. In addition, 2511 MWt is the licensed maximum steady-state power level of the units. This maximum steady-state power level will never knowingly be exceeded.

Conservatism incorporated into the transient analysis is documented in References 1 and 2. Transient analyses are initiated at the conditions given in these References.

The scram delay time and rate of rod insertion allowed by the analyses are conservatively set equal to the longest delay and slowest insertion rate acceptable by technical specifications. The effects of scram worth, scram delay time, and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity is assured by the time requirements for 5% and 20% insertion. By the time the rods are 60% inserted, approximately 4 dollars of negative reactivity have been inserted, which strongly turns the transient and accomplishes the desired effect. The times for 50% and 90% insertion are given to assure proper completion of the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

The MCPR operating limit is, however, adjusted to account for the statistical variation of measured scram times as discussed in Reference 2 and the bases of Specification 3.5.K.

Steady-state operation without forced recirculation will not be permitted except during startup testing. The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculation pumps.

The bases for individual trip settings are discussed in the following paragraphs.

For analyses of the thermal consequences of the transients, the MCPR's stated in the CORE OPERATING LIMITS REPORT as the limiting condition of operation bound those which are conservatively assumed to exist prior to initiation of the transients.

A. Neutron Flux Trip Settings

1. APRM Flux Scram Trip Setting (Run Mode)

   The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of rated thermal power. Because fission chambers provide the basis input signals, the APRM system responds directly to average neutron flux. During transients the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel.

TABLE 3.2-3

INSTRUMENTATION THAT INITIATES ROD BLOCK

Minimum Number of Operable or Tripped Instrument Channels per Trip System (1)	Instrument	Trip Level Setting
2	APRM upscale (flow bias)[7]	$\leq [0.58 W_D + 50] \frac{FRP}{MFLPD}$ [2]
2	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]	[10]
1	Rod block monitor downscale[7]	$\geq$3/125 full scale
3	IRM downscale[3] [8]	$\geq$3/125 full scale
3	IRM upscale[8]	$\leq$108/125 full scale
2[5]	SRM detector not in Startup position [4]	$\geq$2 feet below core centerline
3	IRM detector not in Startup position [8]	$\geq$2 feet below core centerline
2[5] [6]	SRM upscale	$\leq 10^5$ counts/sec
2[5]	SRM downscale [9]	$\geq 10^2$ counts/sec
1 (per bank)	High water level in scram discharge volume (SDV)	$\leq$ 25 gallons (per bank)
1	SDV high water level scram trip bypassed	NA

Notes

1. For the Startup/Hot Standby and Run positions of the reactor mode selector switch, there shall be two operable or tripped trip systems for each function except the SRM rod blocks. IRM upscale and IRM downscale need not be operable in the Run position, APRM downscale, APRM upscale (flow biased), and RBM downscale need not be operable in the Startup/Hot Standby mode. The RBM upscale need not be operable at less than 30% rated thermal power. One channel may be bypassed above 30% rated thermal power provided that a limiting control rod pattern does not exist. For systems with more than one channel per trip system, if the first column cannot be met for one of the two trip systems, this condition may exist for up to 7 days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than 7 days the system shall be tripped. If the first column cannot be met for both trip systems, the systems shall be tripped.

2. W_D is the percent of drive flow required to produce a rated core flow of 98 million lb/hr. Trip level setting is in percent of rated power (2511 MWt).

3. IRM downscale may be bypassed when it is on its lowest range.

4. This function is bypassed when the count rate is $\geq$ 100 cps.

5. One of thé four SRM inputs may be bypassed.

6. This SRM function may be bypassed in the high IRM ranges (ranges 8, 9, and 10) when the IRM upscale rod block is operable.

7. Not required to be operable when performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MWt.

8. This IRM function occurs when the reactor mode switch is in the Refuel or Startup/Hot Standby position.

9. This trip is bypassed when the SRM is fully inserted.

10. The Rod Block Monitor upscale setpoint shall be established as specified in the CORE OPERATING LIMITS REPORT.

sidered inoperable, fully inserted into the core, and electrically disarmed.

5. If the overall average of the 20% insertion scram time data generated to date in the current cycle exceeds the limit specified in the CORE OPERATING LIMITS REPORT, the MCPR operating limit must be modified as required by Specification 3.5.K.

provide reasonable assurance that proper control rod drive performance is being maintained. The results of measurements performed on the control rod drives shall be submitted in the annual operating report to the NRC.

5. The cycle cumulative mean scram time for 20% insertion will be determined immediately following the testing required in Specifications 4.3.C.1 and 4.3.C.2 and the MCPR operating limit adjusted, if necessary, as required by Specification 3.5.K.

D. Control Rod Accumulators

At all reactor operating pressures, a rod accumulator may be inoperable provided that no other control rod in the nine-rod square array around that rod has:

1. An inoperable accumulator,
2. A directional control valve electrically disarmed while in a nonfully inserted position, or
3. A scram insertion greater than maximum permissible insertion time.

If a control rod with an inoperable accumulator is inserted full-in and its directional control valves are electrically disarmed, it shall not be considered to have an inoperable accumulator, and the rod block associated with that inoperable accumulator may be bypassed.

D. Control Rod Accumulators

Once a shift, check the status of the pressure and level alarms for each accumulator.

E. Reactivity Anomalies

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1% Δ k. If this limit is exceeded, the reactor shall be shutdown until the cause has been determined and corrective actions have been taken. In accordance with Specification 6.6, the NRC shall be notified of this reportable occurrence within 24 hours.

E. Reactivity Anomalies

During the startup test program and startups following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every equivalent full power month.

F. Economic Generation Control System

Operation of the unit with the economic generation control system with automatic flow control shall be permissible only in the range of 65% to 100% of rated core flow, with reactor power above 20%.

F. Economic Generation Control System

Prior to entering EGC and once per shift while operating in EGC, the EGC operating parameters will be reviewed for acceptability.

C. Scram Insertion Times

The control rod system is analyzed to bring the reactor subcritical at a rate fast enough to prevent fuel damage, i.e., to prevent the MCPR from becoming less than the fuel cladding integrity safety limit.

Analysis of the limiting power transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above specification, provide the required protection, and MCPR remains greater than the fuel cladding integrity safety limit. It is necessary to raise the MCPR operating limit (per Specification 3.5.K) when the average 20% scram insertion time reaches the limit specified in the CORE OPERATING LIMITS REPORT on a cycle cumulative basis (overall average of surveillance data to date) in order to comply with assumptions in the implementation procedure for the ODYN transient analysis computer code. The basis for choosing this 20% scram insertion time limit is discussed further in the bases for Specification 3.5.K. In the analytical treatment of the transients, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typically observed time delay of about 210 milliseconds. Approximately 90 milliseconds after neutron flux reaches the trip point, the pilot scram valve solenoid deenergizes and 120 milliseconds later the control rod motion is estimated to actually begin. However, 200 milliseconds rather than 120 milliseconds is conservatively assumed for this time interval in the transient analyses and is also included in the allowable scram insertion times specified in Specification 3.3.C.

The scram times for all control rods will be determined at the time of each refueling outage. A representative sample of control rods will be scram tested during the interval of greater than 16 weeks but not more than 32 weeks.

Scram times of new drives are approximately 2.5 to 3 seconds; lower rates of change in scram times following initial plant operation at power are expected. The test schedule provides reasonable assurance of detection of slow drives before system deterioration beyond the limits of Specification 3.3.C. The program was developed on the basis of the statistical approach outlined below and judgment.

The history of drive performance accumulated to date indicates that the 90% insertion times of new and overhauled drives approximate a normal distribution about the mean which tends to become skewed toward longer scram times as operating time is accumulated. The probability of a drive not exceeding the mean 90% insertion time by 0.75 seconds is greater than 0.999 for a normal distribution. The measurement of the scram performance of the drives surrounding a drive exceeding the expected range of scram performance will detect local variations and also provide assurance that local scram time limits are not exceeded. Continued monitoring of other drives exceeding the expected range of scram times provides surveillance of possible anomalous performance.

The numerical values assigned to the predicted scram performance are based on the analysis of the Dresden 2 startup data and of data from other BWR's such as Nine Mile Point and Oyster Creek.

The occurrence of scram times within the limits, but significantly longer than average, should be viewed as an indication of a systematic problem with control rod drives, especially if the number of drives exhibiting such scram times exceeds eight, the allowable number of inoperable rods.

3. If Specification 3.5.H.1 and 2 cannot be met, reactor startup shall not commence or if operating an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

b. During each operating cycle, the following flood protection level switches shall be functionally tested to give the following control room alarms:

   1) turbine building equipment drain sump high level

   2) vault high level

c. The RHR service water vault sump pump discharge check valves outside the vault shall be tested for integrity, using clean demineralized water, at least once per operating cycle.

d. The condenser pit 5-foot trip circuits for each channel shall be checked once a month. A logic system functional test shall be performed during each refueling outage.

I. Average Planar LHGR

During steady-state power operation, the average linear heat generation rate (APLHGR) of all the rods in any fuel assembly, as a function of average planar exposure, at any axial location, shall not exceed the maximum average planar LHGR specified in the CORE OPERATING LIMITS REPORT. If at any time during operation it is determined by normal surveillance that the limiting value for APLHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the APLHGR is not returned in within the prescribed limits within 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.

I. Average Planar LHGR

Daily during steady-state operation above 25% rated thermal power, the average planar LHGR shall be determined.

J. Local LHGR

During steady-state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR specified in the CORE OPERATING LIMITS REPORT. If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within 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.

K. Minimum Critical Power Ratio (MCPR)

During steady-state operation at rated core flow, MCPR shall be equal to or greater than the MCPR limit specified in the CORE OPERATING LIMITS REPORT.

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 specified in the CORE OPERATING LIMITS REPORT. If any time during operation it is determined by normal surveillance 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 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.

J. Local LHGR

Daily during steady-state power operation above 25% of rated thermal power, the local LHGR shall be determined.

K. Minimum Critical Power Ratio (MCPR)

The MCPR shall be determined daily during steady-state power operation above 25% of rated thermal power.

diesel generators. All of these systems have been sized to perform their intended function considering the simultaneous operation of both units.

These technical specifications contain only a single reference to the operability and surveillance requirements for the shared safety-related features of each plant. The level of operability for one unit must be maintained independently of the status of the other. For example, a diesel (1/2 diesel) which is shared between Units 1 and 2 would have to be operable for continuing Unit 1 operation even if Unit 2 were in a cold shutdown condition and needed no diesel power.

Specification 3.5.F.3 provides that should this occur, no work will be performed which could preclude adequate emergency cooling capability being available. Work is prohibited unless it is in accordance with specified procedures which limit the period that the control rod drive housing is open and assures that the worst possible loss of coolant resulting from the work will not result in uncovering the reactor core. Thus, this specification assures adequate core cooling. Specification 3.9 must be consulted to determine other requirements for the diesel generator.

G. Maintenance of Filled Discharge Pipe

If the discharge piping of the core spray, LPCI mode of the RHR, HPCI, and RCIC are not filled, a water hammer can develop in this piping, threatening system damage and thus the availability of emergency cooling systems when the pump and/or pumps are started. An analysis has been done which shows that if a water hammer were to occur at the time emergency cooling was required, the systems would still perform their design function. However to minimize damage to the discharge systems and to ensure added margin in the operation of these systems, this technical specification requires the discharge lines to be filled whenever the system is in an operable condition.

Specification 3.5.F.4 provides assurance that an adequate supply of coolant water is immediately available to the low-pressure core cooling systems and that the core will remain covered in the event of a loss-of-coolant accident while the reactor is depressurized with the head removed.

H. Condensate Pump Room Flood Protection

See Specification 3.5.H

I. Average Planar LHGR

This specification assures that the peak cladding temperature following a postulated design-basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50 Appendix K considering the postulated effects of fuel pellet densification.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average LHGR of all the rods in a fuel assembly at any axial location and is only secondarily dependent on the rod-to-rod power distribution within a fuel assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak cladding temperature by less than ±20°F relative to the peak temperature for a typical fuel design, the limit on the average planar LHGR is sufficient to assure that calculated temperatures are below the 10 CFR 50 Appendix K limit.

The maximum average planar LHGR's specified in the CORE OPERATING LIMITS REPORT are based on calculations employing the models described in Reference 2. Power operation with LHGR's at or below those specified in the CORE OPERATING LIMITS REPORT assures that the peak cladding temperature following a postulated loss-of-coolant accident will not exceed the 2200°F limit. These values represent limits for operation to ensure conformance with 10 CFR 50 and Appendix K only if they are more limiting than other design parameters.

The maximum average planar LHGR's specified in the CORE OPERATING LIMITS REPORT at higher exposures result in a peak cladding temperature of less than 2200°F. However, the maximum average planar LHGR's are specified in the CORE OPERATING LIMITS REPORT as limits because conformance calculations have not been performed to justify operation at LHGR's in excess of those shown.

J. Local LHGR

This specification assures that the maximum linear heat-generation rate in any rod is less than the design linear heat-generation rate specified in the CORE OPERATING LIMITS REPORT even if fuel pellet densification is postulated. The power spike penalty is discussed in Reference 2 and assumes a linearly increasing variation in axial gaps between core bottom and top and assures with 95% confidence that no more than one fuel rod exceeds the design LHGR due to power spiking. No penalty is required in Specification 3.5.L because it has been accounted for in the reload transient analyses by increasing the calculated peak LHGR by 2.2%.

K. Minimum Critical Power Ratio (MCPR)

The steady state values for MCPR specified in the CORE OPERATING LIMITS REPORT 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 plus two percent for uncertainty is satisfied. For any of the special set of transients or disturbances 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 value of MCPR stated in the CORE OPERATING LIMITS REPORT for the limiting condition of operation bounds the initial value of MCPR assumed to exist prior to the initiation of the transients. This initial condition, which is used in the transient analyses, will preclude violation of the fuel cladding integrity safety limit. Assumptions and methods used in calculating the required steady state MCPR limit for each reload cycle are documented in References 2 and 4. The results apply with increased conservatism while operating with MCPR's greater than specified.

The most limiting transients with respect to MCPR are generally:

a) Rod withdrawal error

b) Load rejection or turbine trip without bypass

c) Loss of feedwater heater

Several factors influence which of these transients results in the largest reduction in critical power ratio such as the specific fuel loading, exposure, and fuel type. The current cycle's reload licensing analyses specifies the limiting transients for a given exposure increment for each fuel type. The values specified as the Limiting Condition of Operation are conservatively chosen to bound the most restrictive over the entire cycle for each fuel type.

The need to adjust the MCPR operating limit as a function of scram time arises from the statistical approach used in the implementation of the ODYN computer code for analyzing rapid pressurization events. Generic statistical analyses were performed for plant groupings of similar design which considered the statistical variation in several parameters (initial power level, CRD scram insertion time, and model uncertainty). These analyses (which are described further in Reference 4) produced generic Statistical Adjustment Factors which have been applied to plant and cycle specific ODYN results to yield operating limits which provide a 95% probability with 95% confidence that the limiting pressurization event will not cause MCPR to fall below the fuel cladding integrity safety limit.

As a result of this 95/95 approach, the average 20% insertion scram time must be monitored to assure compliance with the assumed statistical distribution. If the mean value on a cycle cumulative (running average) basis were to exceed a 5% significance level compared to the distribution assumed in the ODYN statistical analyses, the MCPR limit must be increased linearly (as a function of the mean 20% scram time) to a more conservative value which reflects an NRC determined uncertainty penalty of 4.4%. This penalty is applied to the plant specific ODYN results (i.e. without statistical adjustment) for the limiting single failure pressurization event occurring at the limiting point in the cycle. It is not applied in full until the mean of all current cycle 20% scram times reaches the 0.90 secs value of Specification 3.3.C.1. In practice, however, the requirements of 3.3.C.1 would most likely be reached (i.e. individual data set average > .90 secs) and the required actions taken (3.3.C.2) well before the running average exceeds 0.90 secs.

The 5% significance level is defined in Reference 4 as:

$$\tau_B = \mu + 1.65 \left(N_1 / \sum_{i=1}^{n} N_i\right)^{1/2} \sigma$$

where:

μ = Mean value for statistical scram time distribution to 20% inserted

σ = standard deviation of above distribution

N_1 = number of rods tested at BOC (all operable rods)

$\sum_{i=1}^{n} N_i$ = total number of operable rods tested in the current cycle

The value for τ_B used in Specification 3.5.k is specified in the CORE OPERATING LIMITS REPORT and is conservative for the following reasons:

a) For simplicity in formulating and implementing the LCO, a conservative value for $\sum_{i=1}^{n} N_i$ of 708 (i.e. 4x177) was used. This represents one full core data set at BOC plus 6 half core data sets. At the maximum frequency allowed by Specification 4.3.C.2 (16 week intervals) this is equivalent to 24 operating months. That is, a cycle length was assumed which is longer than any past or contemplated refueling interval and the number of rods tested was maximized in order to simplify and conservatively reduce the criteria for the scram time at which MCPR penalization is necessary.

b) The values of μ and σ were also chosen conservatively based on the dropout of the position 39 RPIS switch, since pos. 38.4 is the precise point at which 20% insertion is reached. As a result Specification 3.5.k initiates the linear MCPR penalty at a slightly lower value τ_{ave}. This also produces the full 4.4% penalty at 0.86 secs which would occur sooner than the required value of 0.90 secs.

Local LHGR

The LHGR as a function of core height shall be checked daily during reactor operation at greater than or equal to 25% power to determine if fuel burnup or control rod movement has caused changes in power distribution. A limiting LHGR value is precluded by considerable margin when employing any permissible control rod pattern below 25% rated thermal power.

Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicate that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR.

The daily required for calculating MCPR above 25% rated thermal power is sufficient, since power distribution shifts are very slow when there have not been significant power or control rod changes. In addition, the K_f correction, as specified in the CORE OPERATING LIMITS REPORT, applied to the LCO provides margin for flow increases from low flows.

3. Prior to Single Loop Operation for more than 12 hours, the following restrictions are required:

   a. The MCPR Safety Limit shall be increased by 0.01. (T.S. 1.1A);

   b. The MCPR Operating Limit, as specified in the CORE OPERATING LIMITS REPORT, shall be increased by 0.01 (T.S. 3.5.K);

   c. The flow biased APRM Scram and Rod Block Setpoints shall be reduced by 3.5% to read as follows:

   T.S. 2.1.A.1;
   $S \leq .58WD + 58.5$

   T.S. 2.1.A.1;*
   $S \leq (.58WD + 58.5)$ FRP/MFLPD

   T.S. 2.1.B;
   $S \leq .58WD + 46.5$

   T.S. 2.1.B;*
   $S \leq (.58WD + 46.5)$ FRP/MFLPD

   T.S. 3.2.C (Table 2.1-3);*
   APRM upscale $\leq (.58WD + 46.5)$ FRP/MFLPD

* In the event that MFLPD exceeds FRP.

   d. The flow biased RBM Rod Block setpoints, as specified in the CORE OPERATING LIMITS REPORT, shall be reduced by 4.0%.

   e. The suction valve in the idle loop shall be closed and electrically isolated except when the idle loop is being prepared for return to service.

The licensee's analyses indicate that above 80% power the loop select logic could not be expected to function at a speed differential of 15%. Below 80% power, the loop select logic would not be expected to function at a speed differential of 20%. This specification provides a margin of 5% in pump speed differential before a problem could arise. If the reactor is operating on one pump, the loop select logic trips that pump before making the loop selection.

Analyses have been performed which support indefinite single loop operation provided the appropriate restrictions are implemented within 12 hours. The MCPR Safety Limit has been increased by 0.01 to account for core flow and TIP reading uncertainties which are used in the statistical analysis of the safety limit. The MCPR Operating Limit, as specified in the CORE OPERATING LIMITS REPORT, has also been increased by 0.01 to maintain the same margin to the safety limit as during Dual Loop operation.

The flow biased scram and rod block setpoints are reduced to account for uncertainties associated with backflow through the idle jet pumps when the operating recirculation pump is above 20 - 40% of rated speed. This assures that the flow biased trips and blocks occur at conservative neutron flux levels for a given core flow.

The closure of the suction valve in the idle loop prevents the loss of LPCI flow through the idle recirculation pump into the downcomer.

2. A tabulation shall be submitted on an annual basis of the number of station utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job function (Note: this tabulation supplements the requirements of Section 20.407 of 10 CFR 20), e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

3. Monthly Operating Report

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the Director, Office of Management Information and Program Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555, with a copy to the appropriate Regional Office, to arrive no later than the 15th of each month following the calendar month covered by the report. In addition, any changes to the ODCM shall be submitted with the Monthly Operating Report within 90 days of the effective date of the change.

A report of major change to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted by the onsite review function. If such change is re-evaluated and not installed, notification of cancellation of the change should be provided to the NRC.

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 Rod Withdrawal Block Monitor Upscale Instrumentation Setpoint for Table 3.2-3 of Specification 3.2.C and for Specification 3.6.H.

(2) The overall average of the 20% insertion scram time data for Specification 3.3.C.

(3) The Average Planar Linear Heat Generation Rate (APLHGR) for Specification 3.5.I.

(4) The Linear Heat Generation Rate (LHGR) for Specification 3.5.J.

(5) The Minimum Critical Power Ratio (MCPR) for Specification 3.5.K and 3.6.H.

(6) The K_f core flow MCPR adjustment factor for Specification 3.5.K.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel (latest approved revision).

c. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

d. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 120 TO FACILITY OPERATING LICENSE NO. DPR-29

AND AMENDMENT NO. 116 TO FACILITY OPERATING LICENSE NO. DPR-30

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-254/265

1.0 INTRODUCTION

By letter dated July 11, 1989, as amended by letter dated August 14, 1989, Commonwealth Edison Company (the licensee) proposed changes to the Technical Specifications (TS) for Quad Cities Nuclear Power Station, Units 1 and 2. The proposed changes would modify specifications having cycle-specific parameter limits by replacing these limits with a reference to the Core Operating Limits Report (COLR). The proposed changes would also add the COLR to the Definitions and Administrative Controls Sections of TS. Guidance on the proposed changes was developed by NRC on the basis of a lead-plant proposal submitted on the Oconee plant docket by Duke Power Company. Subsequently, this guidance was provided to all power reactor licensees and applicants by Generic Letter 88-16, dated October 4, 1988.

2.0 EVALUATION

The licensee's proposed changes to the TS are in accordance with the guidance provided by Generic Letter 88-16 and are addressed below.

(1) The Definition Section of the TS was modified to include a definition of the Core Operating Limits Report. This definition notes that plant operation within these limits is addressed by individual specifications.

(2) The following specifications were revised to replace the values of cycle-specific parameter limits with a reference to the COLR that provides these limits.

   a. Specifications 3.2.C (Table 3.2-3) and 3.6.H.3d

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The Rod Withdrawal Block Monitor Upscale Instrumentation Setpoint equation for these Specifications is provided in the COLR.

b. Specification 3.3.C.5

The overall average of the 20 percent insertion scram time limit for this Specification is provided in the COLR.

c. Specification 3.5.I

The Average Planar Linear Heat Generation Rate (APLHGR) limit for this Specification is provided in the COLR.

d. Specification 3.5.J

The Linear Heat Generation Rate (LHGR) limit for this Specification is provided in the COLR.

e. Specifications 3.5.K and 3.6.H.3a

The Minimum Critical Power Ration (MCPR) limits for these Specifications are provided in the COLR.

(f) Specification 3.5.K

The K_f factors that are applied to the operating limit minimum critical power ratio (MCPR) for this Specification are provided in the COLR.

The bases of affected specifications have been modified by the licensee to include appropriate reference to the COLR. Based on our review, we conclude that the changes to these bases are acceptable.

(3) Specification 6.6.A.4 was added to the reporting requirements of the Administrative Controls Section of the TS. This specification requires that the COLR be submitted, upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector. The COLR provides the values of cycle-specific operating limits that are applicable for the current fuel cycle. Furthermore, this specification requires that the values of these limits be established using NRC approved methodology and be consistent with all applicable limits of the safety analysis. The approved methodology is NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (latest approved version). Finally, this specification requires that all changes in cycle-specific parameter limits be documented in the COLR before each reload cycle or remaining part of a reload cycle, and submitted upon issuance to the NRC prior to reactor operation within the new parameter limits.

Based upon a detailed review of the above items, the NRC staff concludes that the licensee provided an acceptable response to Generic Letter 88-16 on removing cycle-specific operating limits from TS. Because plant operation continues to be limited in accordance with the values of cycle-specific limits that are established using an NRC approved methodology, the NRC staff concludes that this change is administrative in nature and there is no impact on plant safety as a consequence. Accordingly, the staff finds that the proposed changes are acceptable.

As part of the implementation of Generic Letter 88-16, the staff also reviewed a sample COLR provided by the licensee. On the basis of this review, the staff concludes that the format and content of the sample COLR are acceptable.

NRC staff reviewed the request by Commonwealth Edison Company to modify the Technical Specifications of the Quad Cities station, Units 1 and 2, to remove the specific values of some cycle-dependent operating limits and place them in a Core Operating Limits Report referenced by TS. NRC staff concludes that the aforementioned Technical Specification modifications are acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change to a requirement with respect to the installation or use of a facility component located within the restricted areas as defined in 10 CFR Part 20. The staff has determined that these amendments involve 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 these amendments involve no significant hazards consideration and there has been no public comment on such finding. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). These amendments also involve changes in recordkeeping, reporting or administrative procedures or requirements. Accordingly, with respect to these items, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR §51.22(c)(10). Pursuant 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

The staff has concluded, based on the considerations discussed previously, 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 this amendment will not be inimical to the common defense and security nor to the health and safety of the public.

Principal Contributor: D. Fieno

Dated: October 20, 1989