February 21, 1991

Docket No. 50-254

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Mr. Thomas J. Kovach
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Dear Mr. Kovach:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. 79288)

The Commission has issued the enclosed Amendment No. 129 to Facility Operating License No. DPR-29 for the Quad Cities Nuclear Power Station, Unit 1. The amendment is in response to your application dated December 18, 1990.

The amendment changes the Technical Specifications to reflect a modification to the fast acting solenoid valves which initiate rapid closure of the turbine control valves. The new design uses a pressure switch, rather than a limit switch, to initiate a reactor scram.

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

Sincerely,

Original Signed By:

Leonard N. Olshan, Project Manager
Project Directorate III/2
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 129 to License No. DPR-29
2. Safety Evaluation

cc w/enclosures:
See next page

DOCUMENT NAME: [AMENDMENT 79288]

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Date:	1/2?/91	1/24/91	2/20/91	1/25/91

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Commonwealth Edison Company

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Unit Nos. 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. 129
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 December 18, 1990, 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 3.B. of Facility Operating License No. DPR-29 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 129 , 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

Richard J. Barrett, Director
Project Directorate III/2
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 21, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 129

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
1.1/2.1-4	1.1/2.1-4
1.1/2.1-15	1.1/2.1-15
3.1/4.1-8	3.1/4.1-8
3.1/4.1-10	3.1/4.1-10
3.1/4.1-13	3.1/4.1-13
3.1/4.1-14	3.1/4.1-14
3.1/4.1-17	3.1/4.1-17

C. Power Transient

1. The neutron flux shall not exceed the scram setting established in Specification 2.1.A for longer than 1.5 seconds as indicated by the process computer.

2. When the process computer is out of service, this safety limit shall be assumed to be exceeded if the neutron flux exceeds the scram setting established by Specification 2.1.A and a control rod scram does not occur.

C. Reactor low water level scram setting shall be 144 inches above the top of the active fuel* at normal operating conditions.

D. Reactor Water Level (Shutdown Condition)

Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than that corresponding to 12 inches above the top of the active fuel* when it is seated in the core.

*Top of active fuel is defined to be 360 inches above vessel zero (See Bases 3.2).

D. Reactor low water level ECCS initiation shall be ≥ 84 inches above the top of the active fuel* at normal operating conditions.

E. Turbine stop valve scram shall be ≤ 10% valve closure from full open.

F. The scram for turbine control valve fast closure due to actuation of the fast acting solenoid valve shall be ≥ 460 psig EHC fluid pressure.

G. Main steamline isolation valve closure scram shall be ≤ 10% valve closure from full open.

*Top of active fuel is defined to be 360 inches above vessel zero (See Bases 3.2).

F. Turbine Control Valve Fast Closure Scram

The turbine control valve fast closure scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection and subsequent failure of the bypass, i.e., it prevents MCPR from becoming less than the MCPR fuel cladding integrity safety limit for this transient. For the load rejection without bypass transient from 100% power, the peak heat flux (and therefore LHGR) increases on the order of 15% which provides wide margin to the value corresponding to 1% plastic strain of the cladding.

The trip setpoint of $\geq$ 460 psig EHC fluid pressure was developed to ensure that the pressure switch is actuated prior to the closure of the turbine control valves (at approximately 400 psig EHC fluid pressure) yet assure that the system is not actuated unnecessarily due to EHC system pressure transients which may cause EHC system pressure to momentarily decrease.

G. Reactor Coolant Low Pressure Initiates Main Steam Isolation Valve Closure

The low pressure isolation at 825 psig was provided to give protection against fast reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs in the Run mode when the main steamline isolation valves are closed to provide for reactor shutdown so that operation at pressures lower than those specified in the thermal hydraulic safety limit does not occur, although operation at a pressure lower than 825 psig would not necessarily constitute an unsafe condition.

H. Main Steamline Isolation Valve Closure Scram

The low pressure isolation of the main steamlines at 825 psig was provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature in the Run mode which occurs when the main steamline isolation valves are closed to provide for reactor shutdown so that high power operation at low reactor pressures does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 825 psig requires that the reactor mode switch be in the Startup position, where protection of the fuel cladding integrity safety limit is provided by the IRM and APRM high neutron flux scrams. Thus, the combination of main steamline low-pressure isolation and isolation valve closure scram in the Run mode assures the availability of neutron flux scram protection over the entire range of applicability of fuel cladding integrity safety limit. In addition, the isolation valve closure scram in the Run mode anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. With the scrams set at 10% valve closure in the Run mode, there is no increase in neutron flux.

To satisfy the long-term objective of maintaining an adequate level of safety throughout the plant lifetime, a minimum goal of 0.9999 at the 95% confidence level is proposed. With the one-out-of-two taken twice logic, this requires that each sensor have an availability of 0.993 at the 95% confidence level. This level of availability may be maintained by adjusting the test interval as a function of the observed failure history (Reference 1). To facilitate the implementation of this technique, Figure 4.1-1 is provided to indicate an appropriate trend in test interval. The procedure is as follows:

1. Like sensors are pooled into one group for the purpose of data acquisition.
2. The factor M is the exposure hours and is equal to the number of sensors in a group, n, times the elapsed time T(M=nT).
3. The accumulated number of unsafe failures is plotted as an ordinate against M as an abscissa on Figure 4.1-1.
4. After a trend is established, the appropriate monthly test interval to satisfy the goal will be the test interval to the left of the plotted points.
5. A test interval of 1 month will be used initially until a trend is established.

The turbine control valve fast acting solenoid valve pressure switches directly measure the trip oil pressure that causes the turbine control valves to close in a rapid manner. The reactor scram setpoint was developed in accordance with NEDC 31336 "General Electric Instrument Setpoint Methodology" dated October, 1986. As part of the calculation, a calibration period is inputted to achieve a nominal trip point and an allowable setpoint (Technical Specification value). The nominal setpoint is procedurally controlled. Based on the calculation input, the calibration period is defined to be every Refueling Outage.

Group 2 devices utilize an analog sensor followed by an amplifier and a bistable trip circuit. The sensor and amplifier are active components, and a failure is almost always accompanied by an alarm and an indication of the source of trouble. In the event of failure, repair or substitution can start immediately. An as-is failure is one that "sticks" midscale and is not capable of going either up or down in response to an out-of-limits input. This type of failure for analog devices is a rare occurrence and is detectable by an operator who observes that one signal does not track the other three. For purposes of analysis, it is assumed that this rare failure will be detected within 2 hours.

The bistable trip circuit which is a part of the Group 2 devices can sustain unsafe failures which are revealed only on test. Therefore, it is necessary to test them periodically.

A study was conducted of the instrumentation channels included in the Group 2 devices to calculate their 'unsafe' failure rates. The analog devices (sensors and amplifiers) are predicted to have an unsafe failure rate of less than 20×10^{-6} failures/hour. The bistable trip circuits are predicted to have an unsafe failure rate of less than 2×10^{-6} failures/hours. Considering the 2-hour monitoring interval for the analog devices as assumed above and a weekly test interval for the bistable trip circuits, the design reliability goal of 0.99999 is attained with ample margin.

Group 3 devices are active only during a given portion of the operation cycle. For example, the IRM is active during startup and inactive during full-power operation. Thus, the only test that is meaningful is the one performed just prior to shutdown or startup, i.e., the tests that are performed just prior to use of the instrument.

Calibration frequency of the instrument channel is divided into two groups. These are as follows:

1. Passive type indicating devices that can be compared with like units on a continuous basis, and
2. Vacuum tube or semiconductor devices and detectors that drift or lose sensitivity.

Experience with passive type instruments in Commonwealth Edison generating stations and substations indicate that the specified calibrations are adequate. For those devices which employ amplifiers, etc. drift specifications call for drift to be less than 0.4%/month i.e., in the period of a month a drift of 0.4% would occur, thus providing for adequate margin.

The sensitivity of LPRM detectors decreases with exposure to neutron flux at a slow and approximately constant rate. Changes in a power distribution and electronic drift also require compensation. This compensation is accomplished by calibrating the APRM system every 7 days using heat balance data by calibrating individual LPRM's at least every 1000 equivalent full-power hours using TIP traverse data. Calibration on this frequency assures plant operation at or below thermal limits.

A comparison of Tables 4.1-1 and 4.1-2 indicates that some instrument channels have not been included in the latter table. These are mode switch in shutdown, manual scram, high water level in scram discharge volume, main steamline isolation valve closure, and turbine stop valve closure. All of the devices or sensors associated with these scram functions are simple on-off switches, hence calibration is not applicable, i.e., the switch is either on or off. Further, these switches are mounted solidly to the device and have a very low probability of moving; e.g., the thermal switches in the scram discharge volume tank. Based on the above, no calibration is required for these instrument channels.

B. The MFLPD shall be checked once per day to determine if the APRM scram requires adjustment. This may normally be done by checking the LPRM readings, TIP traces, or process computer calculations. Only a small number of control rods are moved daily, thus the peaking factors are not expected to change significantly and a daily check of the MFLPD is adequate.

References

1. I. M. Jacobs, "Reliability of Engineered Safety Features as a Function of Testing Frequency", Nuclear Safety, Vol. 9, No. 4, pp. 310-312, July-August 1968,
2. Licensing Topical Report NEDO-21617-A (December 1978).
3. NEDC - 31336 "General Electric Instrument Setpoint Methodology" dated October, 1986

TABLE 3.1-3

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS RUN MODE

Minimum Number of Operable or Tripped Instrument Channels per Trip System[1]	Trip Function	Trip Level Setting	Action[2]
1	Mode switch in shutdown		A
1	Manual scram		A
	APRM[3]		
2	High Flux (flow biased)	Specification 2.1.A.1	A or B
2	Inoperative		A or B
2	Downscale [11]	≥ 3/125 of full scale	A or B
2	High-reactor pressure	≤ 1060 psig	A
2	High drywell pressure	≤ 2.5 psig	A
2	Reactor low water level	≥ 8 inches[8]	A
2 (per bank)	High-water level in scram discharge volume	≤ 40 gallons per bank	A
2	Turbine condenser low vacuum	≥ 21 inches Hg vacuum	A or C
2	Main Steamline high radiation [12]	≤ 15 X normal full power power background (without hydrogen addition)	A or C
4	Main steamline isolation valve closure [6]	≤ 10% valve closure	A or C
2	Turbine control valve fast closure, valve trip system oil pressure low [9]	≥ 460 psig [10]	A or C
2	Turbine stop valve closure [9]	≤ 10% valve closure	A or C
2	Turbine EHC control fluid low pressure [9]	≥ 900 psig	A or C

TABLE 3.1-4

NOTES FOR TABLES 3.1-1, 3.1-2, AND 3.1-3

[1] There shall be two operable trip systems or one operable and one tripped system for each function.

[2] If the first column cannot be met for one of the trip systems, that trip system shall be tripped. If the first column cannot be met for both trip systems, the appropriate actions listed below shall be taken:

A. Initiate insertion of operable rods and complete insertion of all operable rods within 4 hours.

B. Reduce power level to IRM range and place mode switch in the Startup/Hot Standby position within 8 hours.

C. Reduce turbine load and close main steamline isolation valves within 8 hours.

[3] An APRM will be considered inoperable if there are fewer than 2 LPRM inputs per level or there are less than 50% of the normal complement of LPRM's to an APRM.

[4] Permissible to bypass, with control rod block for reactor protection system reset in refuel and shutdown positions of the reactor mode switch.

[5] Not required to be operable when primary containment integrity is not required.

[6] The design permits closure of any one line without a scram being initiated.

[7] Automatically bypassed when reactor pressure is < 1060 psig.

[8] The +8-inch trip point is the water level as measured by the instrumentation outside the shroud. The water level inside the shroud will decrease as power is increased to 100% in comparison to the level outside the shroud to a maximum of 7 inches. This is due to the pressure drop across the steam dryer. Therefore, at 100% power, an indication of +8 inch water level will actually be +1 inch inside the shroud. 1 inch on the water level instrumentation is $\geq$ 504" above vessel zero. (See Bases 3.2).

[9] Permissible to bypass when first stage turbine pressure is less than that which corresponds to 45% rated steam flow. (< 400 psi)

[10] Trip is indicative of turbine control valve fast closure (due to low EHC fluid pressure) as a result of fast acting valve actuation.

[11] The APRM downscale trip function is automatically bypassed when the IRM instrumentation is operable and not high.

[12] Channel shared by the reactor protection and containment isolation system.

TABLE 4.1-2

SCRAM INSTRUMENT CALIBRATION

MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group[1]	Calibration Standard[5]	Minimum Frequency[2]
High flux IRM	C	Comparison to APRM after heat balance	Every controlled shutdown[4]
High flux APRM			
Output signal	B	Heat balance	Once every 7 days
Flow bias	B	Standard pressure and voltage source	Refueling outage
LPRM	B[6]	Using TIP system	Every 1000 equivalent full power hours
High reactor pressure	A	Standard pressure source	Every 3 months
High drywell pressure	A	Standard pressure source	Every 3 months
Reactor low water level	B	Water level	[7]
Turbine condenser low vacuum	A	Standard vacuum source	Every 3 months
Main steamline high radiation	B	Appropriate radiation source [3]	Refueling outage
Turbine EHC control fluid low pressure	A	Pressure source	Every 3 months
Turbine control valve fast closure	A	Pressure source	Refueling outage
Highwater level in scram discharge volume (dp only)	A	Water level	Refueling outage

Notes:

[1] A description of the three groups is included in the bases of this specification.

[2] Calibration tests are not required when the systems are not required to be operable or are tripped. If tests are missed, they shall be performed prior to returning the systems to an operable status.

[3] A current source provides an instrument channel alignment every 3 months.

[4] Maximum calibration frequency need not exceed once per week.

[5] Response time is not part of the routine instrument check and calibration but will be checked every refueling outage.

[6] Does not provide scram function.

[7] Trip units are calibrated monthly concurrently with functional testing. Transmitters are calibrated once per operating cycle.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 129 TO FACILITY OPERATING LICENSE NO. DPR-29

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

QUAD CITIES NUCLEAR POWER STATION, UNIT 1

DOCKET NO. 50-254

1.0 INTRODUCTION

By letter dated December 18, 1990, Commonwealth Edison Company (the licensee) proposed changes to the Technical Specifications (TS) for Quad Cities Nuclear Power Station, Unit 1. Additional clarifying information that did not change the initial proposed no significant hazards consideration determination was submitted by letters dated February 4 and 13, 1991. The proposed changes reflect a modification to the fast acting solenoid valves which initiate rapid closure of the turbine control valves. The new design uses a pressure switch, rather than a limit switch, to initiate a reactor scram.

2.0 DISCUSSION AND EVALUATION

The staff evaluated the licensee's submittal in four distinct areas:

a. Pressure switch vs. limit switch function - the new design must not change the original intent of the function.

b. Setpoint calculation - the calculation has to be developed using well developed standards.

c. Surveillance interval - the calibration interval must be consistent with the design and the setpoint calculation.

d. Technical Specification changes - the proposed Technical Specifications must reflect the new design.

The objective of the turbine control fast acting solenoid valves is to protect the turbine from overspeed when the load is suddenly removed. The logic to determine the load reduction is the load control unit of the Electro-Hydraulic Control System. The load unbalance signal activates relays which send a signal to the turbine fast acting solenoid valves. Actuation of the fast acting solenoid valves inputs to the Reactor Protection System to provide a reactor scram. The objective of the scram is to anticipate the rapid increase in the pressure and neutron flux which may result from the fast closure of the turbine control valves and subsequent failure of the turbine steam bypass valves.

The existing design uses a limit switch off the fast acting solenoid valves to initiate a reactor scram. The limit switch is a simple on-off status indicator that changes status depending on the position of the fast acting solenoid valves. There is no periodic calibration of the on-off devices and there is no setpoint calculations associated with the limit switch. The existing fast acting solenoid valves have failed several times during surveillance testing. Because of these failures, the licensee has decided to replace the valves with new valves as recommended by the NSSS vendor, General Electric. The new fast acting solenoid valves are manufactured by Parker-Hannifan and have been used with good results since 1976. The function of the new solenoid valves remains the same. However, the reactor scram is now initiated by a pressure switch instead of a limit switch.

The new design does introduce a new failure mode due to the tubing which connects the pressure switch to the solenoid valve. Rupture of the tubing would initiate a reactor scram. However, this is a conservative action. The industry use of the pressure switch as input to the Reactor Protection System has proven more reliable than the existing limit switch. Therefore, the function of the pressure switch to replace the existing limit switch is acceptable.

The new fast acting solenoid valves require a determination of the pressure setpoint. General Electric, in its generic instrument setpoint methodology (NEDC-31336, October 1986), addressed the Turbine Control Valve Fast Closure. The licensee has followed the General Electric methodology. The staff is presently reviewing the General Electric generic setpoint methodology but has not completed the review. Consequently, the Quad Cities setpoint calculation has been reviewed on a individual basis. The on-going General Electric methodology evaluation has been found acceptable for this particular setpoint calculation. Accordingly, the Quad Cities calculation was reviewed for consistency with the General Electric generic setpoint calculation.

The pressure switches directly measure the trip oil pressure that causes the turbine control valves to close in a rapid manner. This oil pressure is normally about 1500 to 1600 psig, and the control valve does not start to close until the pressure drops to 400 psig. It is considered possible in normal operations for the pressure to drop to 740 psig due to transients. Therefore, the analytical limit is 400 psig, and the operational limit is 740 psig.

The instrument accuracy of the pressure switch is two percent of full scale. Accuracy is conservatively estimated to be one percent of full range. Full scale is 3000 psig. The instrument drift for a six-month interval is equal to the instrument accuracy. Drift is assumed to be random and calculated to be ±104 psig for an 18-month refueling outage. By letter dated February 4, 1991, the licensee reported the drift that was actually experienced by the same pressure switches at other plants. The data indicate that the drift assumed by the licensee is conservative.

Using the above data and the methodology of NEDC-31336, General Electric performed the pressure setpoint calculation. The setpoint calculation determined the allowable value or Technical Specification value of 460 psig and the nominal trip setpoint of 590 psig. The staff concludes that the setpoint calculation is consistent with the General Electric setpoint methodology and, therefore, is acceptable.

The proposed fast acting solenoid valves are designed for the pressure switch to be actuated within 30 milliseconds of the time the control valves begin to close. This time is consistent with the design values used in the reload licensing calculations to analyze the load reject without bypass valve transient. Verification of the 30 milliseconds actuation will be conducted during post-modification testing. Therefore, this modification does not involve a reduction in the margin of safety as previously determined.

The proposed calibration frequency is every refueling outage. This proposed frequency is consistent with the guidance in NUREG-0123, "General Electric Standard Technical Specifications," Revision 4. This interval is also consistent with Technical Specifications for BWR plants licensed in the 1980s. General Electric used the 18-month interval in the pressure setpoint calculation. This frequency is used by General Electric in its generic setpoint methodology (NEDC-31336). Therefore, the staff concludes that the surveillance interval for the turbine control valve fast closure is acceptable.

The proposed Technical Specification change revises Table 4.1-2 to require that the fast acting solenoid valves pressure switch be calibrated every refueling outage. Page 3.1/4.1-10 of the Technical Specifications is revised to delete the description of the turbine control valve fast closure scram device as a simple on-off switch. Table 3.1-3 and Section 2.1.F are revised to accurately define the trip level setting of the turbine control valve fast closure scram to greater than 460 psig Electro-Hydraulic Control oil pressure. In addition, the appropriate sections to the Bases are provided to reflect the new design of the fast acting solenoid valves. Therefore, the staff concludes that the proposed Technical Specification changes reflect the new design of the turbine control fast acting solenoid valves.

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

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

5.0 CONCLUSION

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

Principal Contributor: J. Ibarra, SICB

Date: February 21, 1991