January 23, 1996

Mr. E. Thomas Boulette, Ph.D
Senior Vice President - Nuclear
Boston Edison Company
Pilgrim Nuclear Power Station
RFD #1 Rocky Hill Road
Plymouth, MA 02360

SUBJECT: ISSUANCE OF AMENDMENT NO. 165 TO FACILITY OPERATING LICENSE NO. DPR-35, PILGRIM NUCLEAR POWER STATION (TAC NO. M92909)

Dear Mr. Boulette:

The Commission has issued the enclosed Amendment No. 165 to Facility Operating License No. DPR-35 for the Pilgrim Nuclear Power Station. This amendment is in response to your application dated July 14, 1995, as supplemented September 12 and December 8, 1995.

This amendment revises the Technical Specifications (TSs) to convert the current percentage-based scram time limits to notch-based limits in TS 3.3.C, updates the bases for the notch-based limits in TS 3/4.3.C, updates the valid range of parameters for the GEXL correlation as applied to GE11 fuel in TS Bases 2.1.1, and modifies the methods used for calculating the operating limit minimum critical power ratio in TS 4.11.C.

A copy of the related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly <u>Federal Register</u> Notice.

Sincerely,

ORIGINAL SIGNED BY:

Ronald B. Eaton, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-293

Enclosures: 1. Amendment No. 165 to License No. DPR-35
2. Safety Evaluation

cc w/encls: See next page

<u>Distribution</u>: See attached sheet

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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Boston Edison Company
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Docket No. 50-293

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2. Safety Evaluation

cc w/encls: See next page

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DATED: January 23, 1996

AMENDMENT NO. 165 TO FACILITY OPERATING LICENSE NO. DPR-35-PILGRIM NUCLEAR POWER STATION

Docket File
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

BOSTON EDISON COMPANY

DOCKET NO. 50-293

PILGRIM NUCLEAR POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 165
License No. DPR-35

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

   A. The application for amendment filed by the Boston Edison Company (the licensee) dated dated July 14, 1995, as supplemented September 12 and December 8, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations;

   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.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel G. McDonald Jr

for Ledyard B. Marsh, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 23, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 165

FACILITY OPERATING LICENSE NO. DPR-35

DOCKET NO. 50-293

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
B2-2	B2-2
3/4.3-4	3/4.3-4
3/4.3-5	3/4.3-5
B3/4.3-6	B3/4.3-6
3/4.11-2	3/4.11-2
3/4.11-3	3/4.11-3

FUEL CLADDING INTEGRITY (2.1.1) (Cont)

Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28 x 10^3 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28 x 10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis, GETAB (1), which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the General Electric Critical Quality (X) - Boiling Length (L), GEXL, correlation.

The GEXL correlation is valid over the range of conditions used in the tests of the data used to develop the correlation. These conditions are:

Pressure:	800 to 1300 psia
Mass Flux:	0.1 to 1.5 Mlb/hr-ft^2
Inlet Subcooling:	0 to 70 Btu/lb
Axial Profile:	1.5 chopped cosine
	1.7 inlet peaked
	1.7 outlet peaked
R-Factor	0.95 to 1.20
Rod Array	9X9 GE 11 array

MINIMUM CRITICAL POWER RATIO (2.1.2)

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not result in damage to BWR fuel rods, the critical power at

(Cont)

3.3 REACTIVITY CONTROL (Cont)

B. Control Rods (Cont)

4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second.

5. The RBM shall be operable as required in Table 3.2.C-1, or control rod withdrawal shall be blocked.

C. Scram Insertion Times

1. Average scram insertion time for all operable control rods from de-energization of the scram pilot valve solenoids to drop out (DO) of Notches 04, 24, 34, and 44 shall be no greater than:

Notch Position	Average Scram Times (seconds)
44 DO	0.508
34 DO	1.252
24 DO	2.016
04 DO	3.578

4.3 REACTIVITY CONTROL (Cont)

B. Control Rods (Cont)

4. Prior to control rod withdrawal for startup or during refueling, verify that at least two source range channels have an observed count rate of at least three counts per second.

C. Scram Insertion Times

1. Following each refueling outage, or after a reactor shutdown that is greater than 120 days, each operable control rod shall be subjected to scram time tests from the fully withdrawn position. If testing is not accomplished with the nuclear system pressure above 950 psig, the measured scram insertion time shall be extrapolated to reactor pressures above 950 psig using previously determined correlations. Testing of all operable control rods shall be completed prior to exceeding 40% rated thermal power.

3.3 REACTIVITY CONTROL (Cont)

C. Scram Insertion Time (Cont)

2. Average scram insertion time for the three fastest operable control rods in each group of four control rods in all two-by-two arrays from de-energization of the scram pilot valve solenoids to dropout (DO) of Notches 04, 24, 34 and 44 shall be no greater than:

Notch Position	Average Scram Time (Seconds)
44 DO	0.538
34 DO	1.327
24 DO	2.137
04 DO	3.793

3. The maximum scram insertion time for any operable control rod from de-energization of the scram pilot valve solenoids to dropout of Notch 04 shall not exceed 7.00 seconds.

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 this rod has a:

1. Inoperable accumulator.

2. Directional control valve electrically disarmed while in a non-fully inserted position.

3. Scram insertion time greater than the 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.

4.3 REACTIVITY CONTROL (Cont)

C. Scram Insertion Time (Cont)

2. Within each 120 days of operation, a minimum of 10% of the control rod drives, on a rotating basis, shall be scram tested as in 4.3.C.1. An evaluation shall be completed every 120 days of operation to provide reasonable assurance that proper performance is being maintained.

D. Control Rod Accumulators

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

BASES:

3/4.3 REACTIVITY CONTROL (Cont)

C. Scram Insertion Times

The control rod system is designed 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 Safety Limit MCPR. 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 Safety Limit MCPR.

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 each cycle as a periodic check against deterioration of the control rod performance.

The limits on scram insertion time presented in Technical Specification 3.3.C include an allowance for the uncertainty in the location of the position indicator probes as well as an allowance for the uncertainty in the control rod positions themselves when dropout of the reed switches occur.

D. Control Rod Accumulators

Requiring no more than one inoperable accumulator in any nine-rod square array is based on a series of XY PDQ-4 quarter core calculations of a cold, clean core. The worst case in a nine-rod withdrawal sequence resulted in a k_{eff} <1.0 - other repeating rod sequences with more rods withdrawn resulted in k_{eff} >1.0. At reactor pressures in excess of 800 psig, even those control rods with inoperable accumulators will be able to meet required scram insertion times due to the action of reactor pressure. In addition, they may be normally inserted using the control-rod-drive hydraulic system. Procedural control will assure that control rods with inoperable accumulators will be spaced in a one-in-nine array rather than grouped together.

LIMITING CONDITIONS FOR OPERATION

3.11 REACTOR FUEL ASSEMBLY (Cont)

B. Linear Heat Generation Rate (LHGR)

During reactor power operation, the LHGR shall not exceed the limits 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 two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

C. Minimum Critical Power Ratio (MCPR)

1. During power operation MCPR shall be ≥ the MCPR operating limit specified in the Core Operating Limits Report. If at 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 two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

SURVEILLANCE REQUIREMENTS

4.11 REACTOR FUEL ASSEMBLY (Cont)

B. Linear Heat Generation Rate (LHGR)

The LHGR as a function of core height shall be checked daily during reactor operation at ≥ 25% rated thermal power.

C. Minimum Critical Power Ratio (MCPR)

1. MCPR shall be determined daily during reactor power operation at > 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.5.

2. The value of τ in Specification 3.11.C.2. shall be equal to 1.0 unless determined from the result of surveillance testing of Specification 4.3.C as follows:

   a) τ is defined as

$$\tau = \frac{\tau_{ave} - \tau_B}{1.252 - \tau_B}$$

LIMITING CONDITIONS FOR OPERATION

3.11 REACTOR FUEL ASSEMBLY (Cont)

C. Minimum Critical Power Ration MCPR (Cont'd)

2. The operating limit MCPR values as a function of the τ are given in Table 3.3.1 of the Core Operating Limits Report where τ is given by specification 4.11.C.2.

SURVEILLANCE REQUIREMENTS

4.11 REACTOR FUEL ASSEMBLY (Cont)

C. Minimum Critical Power Ration MCPR (Cont'd)

b. The average scram time to dropout of Notch 34 is determined as follows:

$$\tau_{ave} = \frac{\sum_{i=1}^{n} N_i \tau_i}{\sum_{i=1}^{n} N_i}$$

Where: an n = number of surveillance tests performed to date in the cycle.

N_i = number of active control rods measured in the i^{th} surveillance test.

τ_i = average scram time to dropout of Notch 34 of all rods measured in the i^{th} surveillance test.

c. The adjusted analysis mean scram time (τ_B) is calculated as follows:

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

Where:

μ = mean of the distribution for average scram insertion time to dropout of Notch 34, 0.937 sec.

N_1 = total number of active control rods at BOC during the first surveillance test.

σ = standard deviation of the distribution for average scram insertion time to the dropout of Notch 34, 0.021 seconds.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 165 TO FACILITY OPERATING LICENSE NO. DPR-35

BOSTON EDISON COMPANY

PILGRIM NUCLEAR POWER STATION

DOCKET NO. 50-293

1.0 INTRODUCTION

By letter dated July 14, 1995, as supplemented by letters dated September 12 and December 8, 1995, Boston Edison Company, the licensee for Pilgrim Nuclear Power Station (PNPS), submitted a request for amendment to the Technical Specifications (TSs) and the Bases for TS Section 2.1.1, "Fuel Cladding Integrity," are revised to define the valid range of conditions for the General Electric Critical Quality (X) - Boiling Length (L) (GEXL) correlation for GE11 9x9 fuel. Section 3.3.C of TS, "Scram Insertion Times," is revised to include notch-based scram time limits and eliminate the current percentage-based limits. The bases for TS Section 3/4.3.C are modified to reflect the conversion to notch-based limits. Surveillance Requirement 4.11.C, "Minimum Critical Power Ratio," is also modified to reflect the use of notch-based scram time information, and the use of the ODYN Option B/GEMINI methodology.

In a request for additional information dated August 15, 1995, the NRC staff asked the licensee questions related to the revised valid range of conditions for the GEXL correlation for GE11 9x9 fuel, calculation of uncertainties and adjustment factors used in notch-based scram time limits and the calculation of operating limit minimum critical power ratio (OLMCPR). Additional information and clarifications were provided by the licensee in the letter dated September 12, 1995. By letter dated December 8, 1995, the licensee submitted a revised TS Bases for TS Section 2.1.1 which contained a more complete list of parameters for the GEXL correlation.

From a systems viewpoint, the staff review focused on whether the revised scram times are at least as conservative as the current percentage-based scram times and are developed using approved methodology, which accounts for uncertainties in the measurement and uses the proper adjustment factors, if required. The licensee was also asked to reexamine the most limiting transients in order to provide assurance that plant parameters do not exceed GEXL limitations.

2.0 EVALUATION

Technical Specification 2.1.1, "Fuel Cladding Integrity," Bases

TS 2.1.1, Bases provides a description of the valid range of parameters for which the GEXL correlation can be used in determining the OLMCPR. The valid range of parameters has been updated to reflect the use of GE11 9x9 fuel in the Pilgrim fuel cycle. The licensee has indicated that the GE11 arrays are the limiting fuel assemblies, and that the cycle 11 operating limit MCPR is based on the GE11 fuel design. The version of the GEXL correlation and valid range of parameters developed for the GE11 fuel design were evaluated during an audit review by the staff. The staff's findings are documented in a letter to General Electric dated March 25, 1992. The purpose of the audit review was to determine whether the GE11 fuel design complies with the criteria provided in Amendment No. 22 of the GESTAR report. The review determined that reference to GEXL for GE11 fuel is acceptable. Therefore, reference to the correlation in TS Bases is acceptable.

The range of parameters that have been modified include reducing the range of inlet subcooling from 0-100 BTU/lb to 0-70 BTU/lb and reducing the range of reactor pressure from 800-1400 psig to 800-1300 psia. The upper limit of 1300 psia for the GE11 critical power correlation was chosen by General Electric to bound pressures expected during periods of minimum CPR for anticipated operational occurrences (AOOs). Local and axial peaking factors were also removed. The licensee has stated that these peaking factors do not limit the applicability of the GEXL correlation. The licensee states that the limiting AOOs for pressurization include the generator load rejection without bypass event and the feedwater controller failure event at maximum demand. For the generator load rejection without bypass event the peak pressure is less than 1210 psia and inlet subcooling is less than 48 BTU/lb in the time frame which the minimum CPR occurs. For the feedwater controller failure event the peak pressure is less than 1190 psia and less than 52 BTU/lb. These values are within the valid range of conditions for the GE11 critical power correlation.

Based on its review, the staff has determined that the changes to TS 2.1.1 Bases are acceptable because the GEXL correlation as applied to GE11 fuel was developed based on criteria previously approved by the NRC staff in Amendment No. 22 of the GESTAR report. The staff conducted an audit review to confirm that the GE11 fuel was in compliance with the criteria. The licensee has also provided assurance that the range of valid conditions for the GEXL correlation will bound the expected range of conditions of Pilgrim Final Safety Analysis Report anticipated operational occurrences.

Technical Specification 3.3.C, "Scram Insertion Times"

The percent-based limits have been replaced with notch-based limits in TS 3.3.C.1 (average scram insertion time for all operable control rods), TS 3.3.C.2 (average scram insertion time for the three fastest operable control rods in each group of four control rods in all two-by-two arrays) and

TS 3.3.C.3 (maximum scram-time limit for control rods to be considered operable). The current insertion percentages are 10%, 30%, 50% and 90% for TS 3.3.C.1 and TS 3.3.C.2. These percentages correspond to non-integer control rod notch positions, so that after measurement of actual scram times, an adjustment factor is used to determine the percent-based limit from measured data taken at integral notch locations. The licensee states that the proposed changes make scram insertion time tests easier by eliminating the need to adjust the measured response. The notch-based limits are taken from General Electric analyses previously used as the basis for the percent-based limits. TS Section 3.3.C.3 is intended to identify control rods with severely degraded scram performance by limiting the scram 90% insertion time for any operable control rod to 7.00 seconds. The 90% limit will be changed to notch 04, which is conservative. Textual information in TSs 3.3.C.2 and 3.3.C.3, indicating that scram timing begins upon de-energization of the scram solenoid pilot valves, has been included for clarity and consistency with TS 3.3.C.1.

The conversion to notch-based scram limits requires that the licensee account for sources of uncertainty and adjustment factors for the notch-based limits. The current percentage-based limits account for uncertainties in the location of the position indicating probes and uncertainty in the position of the control rods when "drop out" of the reed switch occurs. Information provided in the September 12, 1995, RAI response indicates that the notch-based scram times account for these uncertainties. The licensee has added information to TS Bases, section 3/4.3.C detailing inclusion of these uncertainties. The RAI response states that no new adjustment factors are required for the conversion to the notch based limits.

Based on its review, the staff has determined that the licensee's conversion to the notch-based scram times is acceptable because the limits are based on General Electric analyses previously used as the basis for the percent-based limits. The text added for clarification in TSs 3.3.C.2 and 3.3.C.3 does not represent a change from current scram time testing methods and is acceptable. The change in TS 3.3.C.3 from using a 7.0 second limit at 90% insertion to a 7.0 second limit at notch 04 is conservative and is acceptable.

Technical Specification 4.11.C, "Minimum Critical Power Ratio"

The changes to TS 4.11.C reflect the licensee's use of the GEMINI methodology to determine the operating limit MCPR. The GEMINI methodology was previously approved by the staff. Within TS 4.11.C, the equation used for calculation of τ has been updated by replacing the 30% insertion limit of 1.275s with the proposed notch 34 limit of 1.252s. The definition of τ_{ave}, τ_i, μ and σ also have been updated to reflect conversion from the 30% insertion to dropout of notch 34. The GEMINI methodology is used in the calculation of the μ and σ values.

Based on its review, the changes made to TS 4.11.C are acceptable. The change in the calculation of τ is consistent with the conversion to notch-based limits. The change in the definition of τ_{ave}, τ_i, μ and σ is also consistent with the conversion to notch-based limits. The previously approved GEMINI methodology is used in the calculation of the revised μ and σ values and is acceptable.

3.0 TECHNICAL EVALUATION CONCLUSION

TS 2.1.1, Bases has been changed to reflect the use of GE11 9x9 fuel. The GEXL correlation as applied to GE11 fuel was developed based on criteria previously approved by the NRC staff in Amendment No. 22 of the GESTAR report. The staff conducted an audit review to confirm that the GE11 fuel was in compliance with the criteria. The licensee has also provided assurance that the range of anticipated conditions for AOOs will be bounded by the range of valid conditions for the GEXL correlation. On these bases, the changes to TS Bases 2.1.1 are acceptable.

TSs 3.3.C.1, 3.3.C.2 and 3.3.C.3 were updated to reflect the use of notch-based scram time limits. The notch-based limits are taken from GE analysis which is used as the basis for the current percent-based limits. Text added for clarification in TSs 3.3.C.2 and 3.3.C.3 does not represent a change from current scram time testing methods. The change in TS 3.3.C.3 from using a 7.0 second limit at 90% insertion to a 7.0 second limit at notch 04 is a conservative change, and is consistent with the conversion to notch-based limits. On these bases, the changes to TSs 3.3.C.1, 3.3.C.2, 3.3.C.3 and 3/4.3.C are acceptable.

TS 4.11.C has been updated to reflect the use of the GEMINI methodology in the calculation of the operating limit MCPR. The changes are consistent with the incorporation of the notch-based limits, and are consistent with the approved methodology. On these bases, the changes to TS 4.11.C are acceptable.

4.0 REFERENCES

1. NEDE-24011-P-A-10 and NEDE-24011-P-A-10-US, "General Electric Standard Application for Reactor Fuel," February and March 1991.

2. Letter to J. S. Charnley from G. C. Lainas dated March 22, 1986; Subject: Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A, Rev. 6, Amendment 11, "General Electric Standard Application for Reactor Fuel" (GESTAR II)

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. 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 (60 FR 39433). 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.

7.0 CONCLUSION

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

Principal Contributor: J. Golub

Date: January 23, 1996