April 21, 1987

Docket No. 50-336

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Mr. Edward J. Mroczka, Senior Vice President
Nuclear Engineering and Operations
Northeast Nuclear Energy Company
P. O. Box 270
Hartford, Connecticut 06141-0270

Dear Mr. Mroczka:

The Commission has issued the enclosed Amendment No. 116 to Facility Operating License No. DPR-65 for Millstone Nuclear Power Station, Unit No. 2, in response to your application dated February 6, 1987.

The change modifies the Technical Specifications (TS) as follows: (1) a new Limiting Condition for Operation (LCO) and a corresponding Surveillance Requirement (SR), TS 3/4.1.7 "Control Rod Drive Mechanisms," assures that control rods cannot be withdrawn prior to establishing conditions consistent with the safety analysis, and (2) a change to the LCO and SR for the reactor protection system (RPS), TS 3/4.3.1, extends operability and surveillance requirements, for the Power Level-High trip function, to Mode 3.

A copy of our Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's next bi-weekly Federal Register notice.

Sincerely,

/S/

David H. Jaffe, Project Manager
Project Directorate I-4
Division of Reactor Projects I/II

Enclosures:
1. Amendment No. 116 to DPR-65
2. Safety Evaluation

cc w/enclosures:
See next page

*See previous white for concurrences - name change only

PD#8*	PD#8*	FOB*	OGC*	PD#-8*	PD I-4
PKreutzer	DJaffe	WRegan	MKarman	AThadani	JStolz
3/31/87	3/31/87	4/1/87	4/3/87	4/7/87	4/21/87

April 21, 1987

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P. O. Box 270
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Sincerely,

/S/

David H. Jaffe, Project Manager
PWR Project Directorate #8
Division of PWR Licensing-B

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See next page

PBD#8	PBD#8	RDB	OGC	PBD#8
PKreutzer	DJaffe	WRegan		AThadani
3/31/87	3/31/87	4/1/87	4/3/87	4/7/87

Mr. Edward J. Mroczka
Northeast Nuclear Energy Company

Millstone Nuclear Power Station
Unit No. 2

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

NORTHEAST NUCLEAR ENERGY COMPANY

THE CONNECTICUT LIGHT AND POWER COMPANY

THE WESTERN MASSACHUSETTS ELECTRIC COMPANY

DOCKET NO. 50-336

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 116
License No. DPR-65

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

   A. The application for amendment by Northeast Nuclear Energy Company, et al. (the licensee), dated February 6, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-65 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, 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 issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
Project Directorate I-4
Division Reactor Projects I/II

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: April 21, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 21

FACILITY OPERATING LICENSE NO. DPR-65

DOCKET NO. 50-336

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain a vertical line indicating the area of change. The corresponding overleaf pages are provided to maintain document completeness.

Remove Pages	Insert Pages
IV	IV
--	3/4 1-31
3/4 3-2	3/4 3-2
3/4 3-4	3/4 3-4
3/4 3-7	3/4 3-7
B 3/4 1-3	B 3/4 1-3
B 3/4 1-5	B 3/4 1-5

INDEX

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITION FOR OPERATION

3.1.3.7 The control rod drive mechanisms shall be de-energized.

APPLICABILITY: MODES 3*, 4, 5 and 6, whenever the RCS boron concentration is less than refueling concentration of Specification 3.9.1.

ACTION:

With any of the control rod drive mechanisms energized, restore the mechanisms to their de-energized state within 2 hours or immediately open the reactor trip circuit breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.7 The control rod drive mechanisms shall be verified to be de-energized at least once per 24 hours.

* The control rod drive mechanisms may be energized for MODE 3 as long as 4 reactor coolant pumps are OPERATING, the reactor coolant system temperature is greater than 500°F, the pressurizer pressure is greater than 2000 psia and the high power trip is operable.

3/4.3.1 REACTOR PROTECTIVE INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the reactor protective instrumentation channels and bypasses of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1.1 Each reactor protective instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-1.

4.3.1.1.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.1.1.3 The REACTOR TRIP SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

4.3.1.1.4 The response time of all REACTOR TRIP SYSTEM resistance temperature detectors (RTD) shall be verified to be less than or equal to the value specified in Table 3.3-2 within one month of operation for newly installed RTD's and once every 18 months thereafter.

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2 and *	1
2. Power Level - High	4	2 (f)	3	1, 2, 3(d)	2
3. Reactor Coolant Flow - Low	4	2(a)	3	1, 2 (e)	2
4. Pressurizer Pressure - High	4	2	3	1, 2	2
5. Containment Pressure - High	4	2	3	1, 2	2
6. Steam Generator Pressure - Low	4	2(b)	3	1, 2	2
7. Steam Generator Water Level - Low	4	2	3	1, 2	2
8. Local Power Density - High	4	2(c)	3	1	2
9. Thermal Margin/Low Pressure	4	2(a)	3	1, 2 (e)	2
10. Loss of Turbine--Hydraulic Fluid Pressure - Low	4	2(c)	3	1	3

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Wide Range Logarithmic Neutron Flux Monitor - Shutdown	4	0	2	3, 4, 5	4
12. Underspeed - Reactor Coolant Pumps	4	2(a)	3	1, 2(e)	2

TABLE 3.3-1 (Continued)

TABLE NOTATION

* With the protective system trip breakers in the closed position and the CEA drive system capable of CEA withdrawal.

(a) Trip may be bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 5% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 600 psia; bypass shall be automatically removed at or above 600 psia.

(c) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 15% of RATED THERMAL POWER.

(d) Trip does not need to be OPERABLE if all the control rod drive mechanisms are de-energized or if the RCS boron concentration is greater than or equal to the refueling concentration of Specification 3.9.1.

(e) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(f) ΔT Power input to trip may be bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 5% of RATED THERMAL POWER.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 4 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels and with the THERMAL POWER level:

a. $<$ 5% of RATED THERMAL POWER, immediately place the inoperable channel in the bypassed condition; restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 5% of RATED THERMAL POWER.

b. $>$ 5% of RATED THERMAL POWER, operation may continue with the inoperable channel in the bypassed condition, provided the following conditions are satisfied:

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	S/U(1)	N.A.
2. Power Level - High				
a. Nuclear Power	S	D(2), M(3),Q	M	1, 2, 3*
b. ΔT Power	S	D(4), Q	M	1
3. Reactor Coolant Flow - Low	S	R	M	1, 2
4. Pressurizer Pressure - High	S	R	M	1, 2
5. Containment Pressure - High	S	R	M	1, 2
6. Steam Generator Pressure - Low	S	R	M	1, 2
7. Steam Generator Water Level - Low	S	R	M	1, 2
8. Local Power Density - High	S	R	M	1
9. Thermal Margin/Low Pressure	S	R	M	1, 2
10. Loss of Turbine--Hydraulic Fluid Pressure - Low	N.A.	N.A.	S/U(1)	N.A.

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
11. Wide Range Logarithmic Neutron Flux Monitor	S	N.A.	S/U(1)	3, 4, 5 and *
12. Underspeed - Reactor Coolant Pumps	S	R	M	1, 2
13. Reactor Protection System Logic	N.A.	N.A.	M and S/U(1)	1, 2
14. Reactor Trip Breakers	N.A.	N.A.	M	1, 2 and *

3/4.1.2 BORATION SYSTEMS (Continued)

The boron capability required below 200°F is based upon providing a 2% $\Delta k/k$ SHUTDOWN MARGIN at 140°F during refueling with all full and part length control rods withdrawn. This condition requires either 5,050 gallons of 6.25% boric acid solution from the boric acid tanks or 57,000 gallons of 1720 ppm borated water from the refueling water storage tank.

A minimum boron concentration of 1720 ppm is required in the RWST at all times in order to satisfy safety analysis assumptions for boron dilution incidents and other transients using the RWST as a borated water source.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of a CEA ejection accident and an uncontrolled CEA withdrawal from subcriticality are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met.

The ACTION statements applicable to an immovable or untrippable CEA and to a large misalignment ($\geq$ 20 steps) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of an immovable or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments ($<$ 20 steps) of the CEAs, there is 1) a small degradation in the peaking factors relative to those assumed in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 2) a small effect on the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 3) a small effect on the available SHUTDOWN MARGIN, and 4) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the small misalignment of a CEA permits a one hour time interval during which attempts may be made to restore the CEA to within its alignment requirements prior to initiating a reduction in THERMAL POWER. The one hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs and (3) minimize the effects of xenon redistribution.

Overpower margin is provided to protect the core in the event of a large misalignment ($\geq$ 20 steps) of a CEA. However, this misalignment would cause distortion of the core power distribution. The reactor

3/4.1.3 MOVABLE CONTROL ASSEMBLIES (Continued)

protective system would not detect the degradation in radial peaking factors and since variations in other system parameters (e.g., pressure and coolant temperature) may not be sufficient to cause trips, it is possible that the reactor could be operating with process variables less conservative than those assumed in generating LCO and LSSS setpoints. Therefore, the ACTION statement associated with the large misalignment of a CEA requires a prompt and significant reduction in THERMAL POWER prior to attempting realignment of the misaligned CEA.

The ACTION statements applicable to misaligned or inoperable CEAs include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements bring the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in 1) local burnup, 2) peaking factors and 3) available shutdown margin which are more adverse than the conditions assumed to exist in the safety analyses and LCO and LSSS setpoints determination. Therefore, time limits have been imposed on operation with inoperable CEAs to preclude such adverse conditions from developing.

Operability of the CEA position indicators (Specification 3.1.3.3) is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits and ensures proper operation of the rod block circuit. The CEA "Full In" and "Full Out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. Therefore, the ACTION statements applicable to inoperable CEA position indicators permit continued operations when the positions of CEAs with inoperable position indicators can be verified by the "Full In" or "Full Out" limits.

CEA positions and OPERABILITY of the CEA position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

The maximum CEA drop time permitted by Specification 3.1.3.4 is the assumed CEA drop time used in the accident analyses. Measurement with $T_{avg} \geq 515°F$ and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES (Continued)

The LSSS setpoints and the power distribution LCOs were generated based upon a core burnup which would be achieved with the core operating in an essentially unrodded configuration. Therefore, the CEA insertion limit specifications require that during MODES 1 and 2, the full length CEAs be nearly fully withdrawn. The amount of CEA insertion permitted by the Long Term Steady State Insertion Limits of Specification 3.1.3.6 will not have a significant effect upon the unrodded burnup assumption but will still provide sufficient reactivity control. The Transient Insertion Limits of Specification 3.1.3.6 are provided to ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of a CEA ejection accident are limited to acceptable levels; however, long term operation at these insertion limits could have adverse effects on core power distribution during subsequent operation in an unrodded configuration.

The control rod drive mechanism requirement of Specification 3.1.3.7 is provided to assure that the consequences of an uncontrolled CEA withdrawal from subcritical transient will stay within acceptable levels. This specification assures that reactor coolant system conditions exist which are consistent with the plant safety analysis prior to energizing the control rod drive mechanisms. The accident is precluded when conditions exist which are inconsistent with the safety analysis since deenergized drive mechanisms cannot withdraw a CEA. The drive mechanisms may be energized with the boron concentration greater than or equal to the refueling concentration since, under these conditions, adequate SHUTDOWN MARGIN is maintained, even if all CEAs are fully withdrawn from the core.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 116 TO DPR-65

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL.

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 2

DOCKET NO. 50-336

INTRODUCTION

By application for license amendment dated February 6, 1987, Northeast Nuclear Energy Company, et al. (the licensee), requested changes to the Technical Specifications (TS) for Millstone, Unit 2 as follows: (1) a new Limiting Condition for Operation (LCO) and a corresponding Surveillance Requirement (SR), TS 3/4.1.7 "Control Rod Drive Mechanisms," has been proposed to assure that control rods cannot be withdrawn prior to establishing conditions consistent with the safety analysis, and (2) a change to the LCO and SR for the reactor protection system (RPS), TS 3/4.3.1, would extend operability and surveillance requirements, for the Power Level-High trip function, to Mode 3.

DISCUSSION AND EVALUATION

On July 23, 1986, the NRC issued Generic Letter (GL) 86-13, "Potential Inconsistency Between Safety Analyses and Technical Specifications." The conclusions of GL 86-13 indicate that the TS may not provide sufficient restrictions to assure that, should a continuous control rod bank withdrawal occur from subcritical conditions, the consequences are within those predicted by the safety analysis. This conclusion is based upon a comparison of Westinghouse safety analysis and plant TS which show that fewer than a full complement of reactor coolant pumps (RCP's) are permitted to be operating at zero power while the safety analysis assumes that all RCP's are operable. Under such conditions, the departure from nucleate boiling ratio (DNBR) criteria demonstrated in the safety analysis might not be met in the event of an accident.

The licensee's letter dated November 4, 1986 submitted LER 86-010-00 which reported that the conclusions of GL 86-13 were valid for Millstone Unit 2. The licensee committed to provide an administrative control for the control element drive mechanisms (CEDM's) to assure that they are de-energized when less than four (all) RCP's are operating; thus, a continuous control bank withdrawal from subcritical conditions would be prevented. The licensee subsequently submitted their application, dated February 6, 1987, which provided proposed changes to the TS to assure that the consequences of any continuous control bank withdrawal from subcritical conditions will be less severe than those predicted by the safety analysis.

The licensee has proposed a new LCO and an associated SR, which reflect a revised evaluation of continuous rod withdrawal from subcritical conditions, to be designated TS 3/4.1.3.7, "Control Rod Drive Mechanisms." The proposed LCO would require the control rod drive mechanisms (CRDMs) to be de-energized in reactor operating Modes 3, 4, 5 and 6 (hot standby and all shutdown modes) unless the reactor coolant system soluble boron concentration exceeds 1720 ppm. A boron concentration of 1720 ppm provides sufficient shutdown margin to prevent criticality in the event of a continuous rod withdrawal. In Mode 3 (hot standby) the CRDM's may be reenergized if "...4 reactor coolant pumps are OPERATING, the reactor coolant system temperature is greater than 500°F, the pressurizer pressure is greater than 2000 psia and the high power trip is operable." The preceding conditions are necessary in order to prevent the DNBR from being less than 1.30 in accordance with the licensee's revised evaluation of continuous rod withdrawal from subcritical conditions. In the event that LCO 3.1.3.7 is not satisfied, the associated Action Statement requires that the reactor trip breakers be opened within 2 hours; this action would also prevent rod withdrawal. The proposed SR requires that the CRDM's be verified to be de-energized at least once per 24 hours.

The proposed changes to the LCO and SR for the RPS, TS 3/4.3.1 are consistent with the revised evaluation of continuous rod withdrawal from subcritical conditions. The Power Level-High trip function, which is the subject of the proposed change, is relied upon to terminate the transient by tripping the reactor. The Power Level-High function would now be required to be operable, and undergo surveillance, in Modes 1,2 and 3; previously these requirements only applied to Modes 1 and 2. The licensee has proposed exceptions to operability requirements for the Power Level-High trip function in Mode 3, as follows:

> Trip does not need to be operable if all the control rod drive mechanisms are de-energized or if the RCS boron concentration is greater than or equal to the refueling concentration of Specification 3.9.1.

Thus, the Power Level-High trip need not be operable when continuous rod withdrawal is precluded. The Power Level-High trip function is not relied upon to terminate any other accidents or transients initiated from Mode 3.

The licensee has also proposed suitable changes to the TS Bases that are consistent with the proposed TS changes.

The proposed changes to the TS are consistent with the licensee's revised evaluation of the continuous control rod withdrawal from subcritical conditions and assure that, should this incident occur, the consequences would be acceptable. In the case of the continuous rod withdrawal from subcritical conditions, a DNBR of greater than or equal to 1.30 assures continued fuel integrity. Accordingly, the proposed change to the TS are acceptable.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or a change in surveillance requirements. The 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 published 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.

CONCLUSION

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

Date: April 21, 1987

Principal Contributor:
D. Jaffe