June 27, 1985

Docket No. 50-316

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Mr. John Dolan, Vice President
Indiana and Michigan Electric Company
c/o American Electric Power Service Corporation
1 Riverside Plaza
Columbus, Ohio 43216

Dear Mr. Dolan:

The Commission has issued the enclosed Amendment No. 69 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated February 14, 1985. The corresponding Unit 2 Techical Specifications change proposed in that letter will be handled by separate correspondence.

The amendment revises the Technical Specifications by changes to the heatup and cooldown curves to reflect the most current reactor vessel material surveillance capsule analysis.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular bi-weekly Federal Register notice.

Sincerely,

/s/DWigginton

David L. Wigginton, Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 69 to DPR-74
2. Safety Evaluation

cc: w/enclosures
See next page

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Donald C. Cook Nuclear Plant

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

INDIANA AND MICHIGAN ELECTRIC COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 69
License No. DPR-74

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

   A. The application for amendment by Indiana and Michigan Electric Company (the licensee) dated February 14, 1985, 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-74 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 69 , 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

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 27, 1985

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 69 FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 4-25	3/4 4-25
3/4 4-26	3/4 4-26
B3/4 4-6	B3/4 4-6

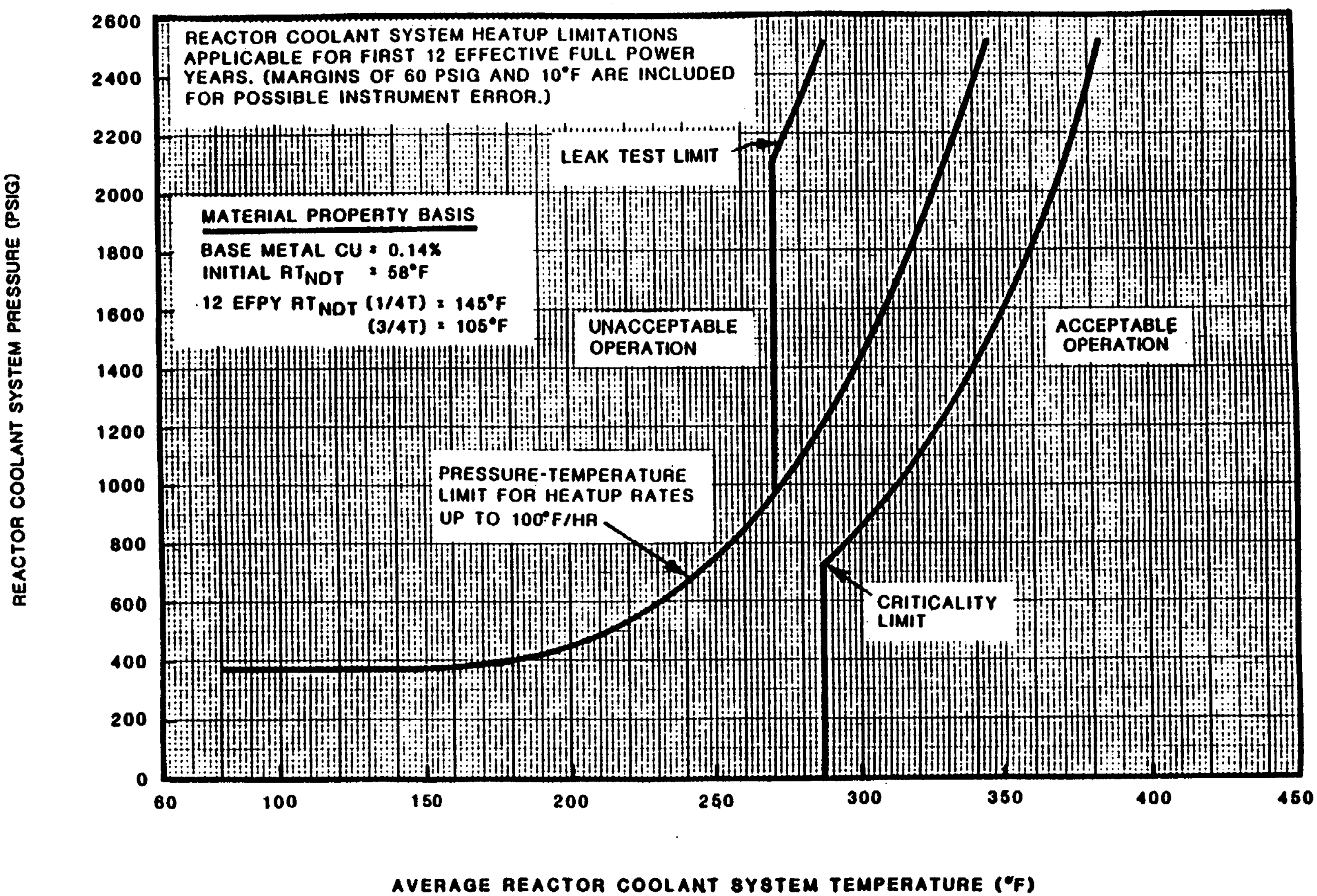

FIGURE 3.4-2

REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS VERSUS 100°F/HR RATE, CRITICALITY LIMIT AND HYDROSTATIC TEST LIMIT

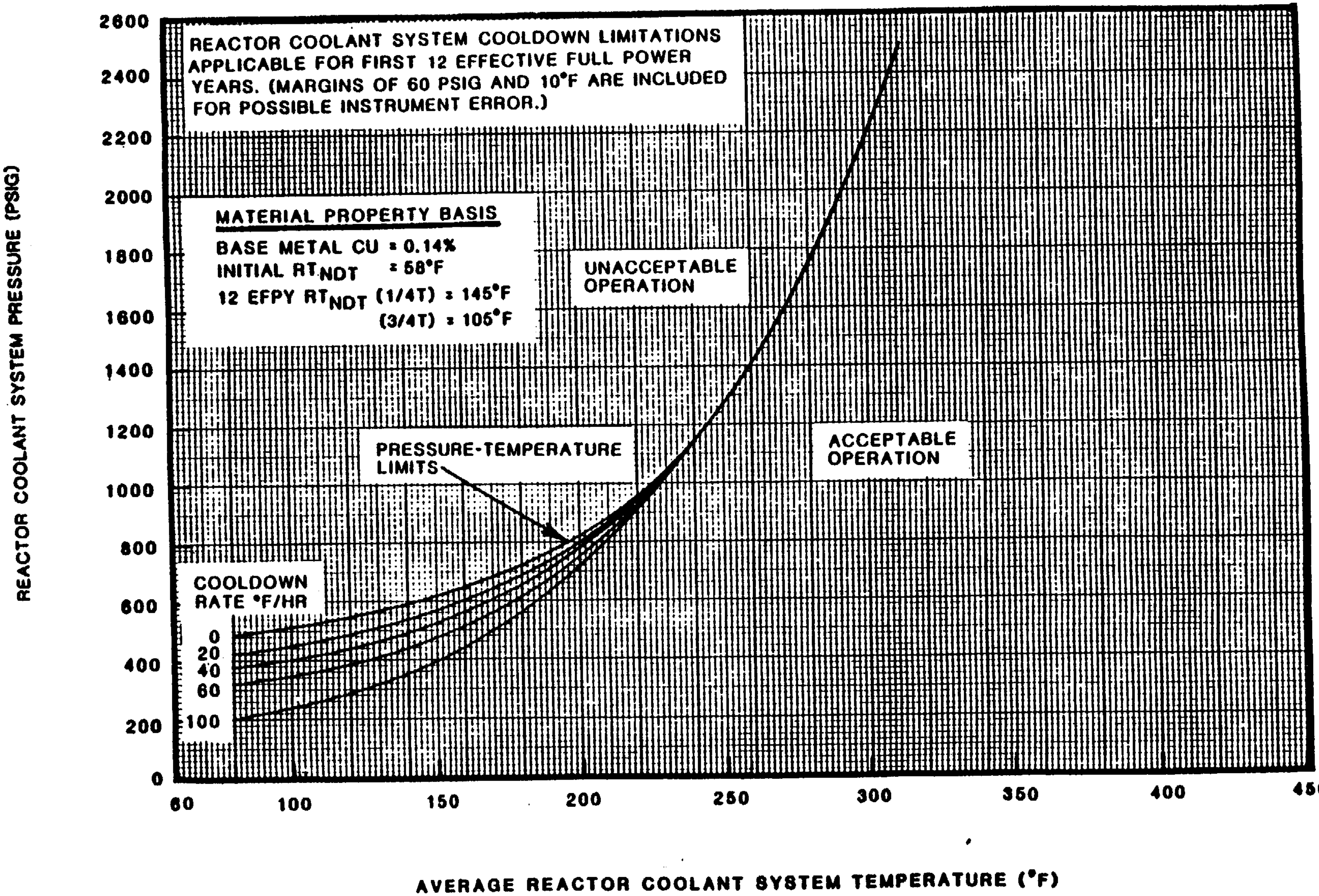

FIGURE 3.4-3

REACTOR COOLANT SYSTEM PRESSURE-TEMPERATURE LIMITS VERSUS COOLDOWN RATES

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4.1.4 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

An ID or OD one-quarter thickness surface flaw is postulated at the location in the vessel which is found to be the limiting case. There are several factors which influence the postulated location. The thermal induced bending stress during heatup is compressive on the inner surface while tensile on the outer surface of the vessel wall. During cooldown the bending stress profile is reversed. In addition, the material toughness is dependent upon irradiation and temperature and therefore the fluence profile through the reactor vessel wall, the rate of heatup and also the rate of cooldown influence the postulated flaw location.

The heatup limit curve, Figure 3.4-2, is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 100°F per hour. The cooldown limit curves of Figure 3.4-3 are composite curves which were prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 12 EFPY.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E > 1 Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence and copper content of the material in question, can be predicted using Figures B 3/4.4-1 and B 3/4.4-2. The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 12 EFPY, as well as adjustments for possible errors in the pressure and temperature sensing instruments. The heatup and cooldown curves are applicable to low leakage cores.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 69 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA AND MICHIGAN ELECTIC COMPANY

DONALD C. COOK NUCLEAR PLANT UNIT NO. 2

DOCKET NO. 50-316

Introduction

In a letter from R. F. Hering to H. R. Denton dated February 14, 1985, the Indiana & Michigan Electric Company requested an amendment to the DCCNP-2 Technical Specifications. The amendment proposes revised reactor coolant pressure-temperature limits, which will be applicable through twelve (12) effective full power years (EFPY). In support of this amendment, the licensee referenced a Southwest Research Institute report entitled, "Reactor Vessel Material Surveillance Program for Donald C. Cook Unit No. 2 Analysis of Capsule Y." This document was transmitted to the staff in a letter from M. P. Alexich to H. R. Denton dated July 20, 1984.

Discussion

Pressure-temperature limits must be calculated in accordance with the requirements of Appendix G, 10 CFR 50, which became effective on July 26, 1983. Pressure-temperature limits that are calculated in accordance with the requirements of Appendix G, 10 CFR 50 are dependent upon the initial RT_{NDT} for the limiting materials in the beltline and closure flange regions of the reactor vessel and the increase in RT_{NDT} resulting from neutron irradiation damage to the limiting beltline material.

The DCCNP-2 reactor vessel was procured to ASME Code requirements, which did not specify fracture toughness testing to determine the initial RT_{NDT} for each reactor vessel material. Technical Specification Table B 3/4.4-1 reports the initial RT_{NDT} for materials in the closure flange and beltline regions of the DCCNP-2 vessel using the method recommended by the staff in Standard Review Plan, Section 5.3.2, Branch Technical Position MTEB 5-2 entitled, "Fracture Toughness Requirements." This method results in an initial RT_{NDT} for the limiting closure flange region material of 33°F, an initial RT_{NDT} for the limiting beltline weld metal of -35°F and an initial RT_{NDT} for the limiting beltline plate of 58°F.

The method recommended by the staff for predicting the increase in RT_{NDT} resulting from neutron irradiation damage is documented in Regulatory Guide 1.99, Rev. 1, April 1977, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." This method of predicting neutron irradiation damage is dependent upon the predicted amount of neutron fluence and the amounts of copper and phosphorus in the beltline material. The predicted amount of neutron fluence is dependent upon the neutron flux. The neutron flux is dependent upon the core design. The DCCNP-2 core design was changed to a low leakage core following the second fuel cycle. The licensee plans to utilize low leakage cores for the remaining life of DCCNP-2 plant. Using flux wire measurements and a two dimensional discrete ordinate transport calculation, the Capsule Y Test Report indicates that the peak inside surface neutron flux during core cycle 3 (low leakage core) is calculated to be 1.59 X 10^{10} n/cm^2/sec (E >1MeV) and the peak inside surface neutron flux during core cycles 1 and 2 is calculated to be 1.98 X 10^{10} n/cm^2/sec (E >1MeV). This results in a predicted neutron fluence for twelve EFPY of 3.8 x 10^{18} n/cm^2 (E >1MeV) at the 1/4 T beltline location and 9.4 X 10^{17} n/cm^2 (E >1MeV) at the 3/4 T beltline location.

The amounts of copper and phosphorus in the beltline materials in the DCCNP-2 reactor vessel are reported in FSAR Appendix Q, Question 121.2. Using the method recommended in Regulatory Guide 1.99, Rev. 1 for predicting neutron irradiation damage, the limiting material in the DCCNP-2 reactor vessel would be plate heat no. C5556-2. The material used in the reactor vessel surveillance program is from plate heat no. C5521-2. Both plates heat no. C5556-2 and C5521-2 were supplied by Lukens Steel, have been heat treated to an equivalent microstructure and have equivalent chemical composition. Hence, the test results from the surveillance material could be used to demonstrate the effect that neutron irradiation would have on the limiting beltline material. In Table 1 we have compared the amount of increase in RT_{NDT} resulting from neutron irradiation damage observed on capsule material from plate heat no. C5521-2 to that predicted by the formula in Regulatory Guide 1.99, Rev. 1. This comparison indicates that at low neutron fluence, the formula in the guide is nonconservative. However, in instances such as this, the guide recommends that neutron irradiation damage be estimated by a straight-line interpolation on a logarithmic plot between credible surveillance points and extrapolation from the lower surveillance data point, using the slope of the family of lines in Figure 1 of the guide. This is the method used by the licensee to predict neutron irradiation damage to the limiting beltline material.

Evaluation/Conclusion

The staff has used the method of calculating pressure-temperature limits in USNRC Standard Review Plan 5.3.2, NUREG-0800, Rev. 1, July 1981 to evaluate the proposed pressure-temperature limits. The amount of neutron irradiation damage to the limiting beltline material was estimated using the extrapolation method recommended in Regulatory Guide 1.99, Rev. 1. Our conclusion is that the proposed pressure temperature limits meet the safety margins of Appendix G, 10 CFR 50 for (12) twelve EFPY and may be incorporated into the DCCNP-2 technical specifications.

TABLE I

INCREASE IN REFERENCE TEMPERATURE, RT_{NDT}, FOR SURVEILLANCE MATERIAL HEAT NO. C5521-2

Neutron Fluence (n/cm^2, E >1 MeV)	Increase in RT_{NDT} (°F) Observed from Capsule Test Data	Predicted by Regulatory Guide 1.99, Rev. 1
2.7×10^{18}	80	65
7.0×10^{18}	100	105

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. 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 issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 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 this 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: June 27, 1985

Principal Contributors:

B. Elliot