May 22, 1991

Docket No. 50-298

Mr. Guy R. Horn
Nuclear Power Group Manager
Nebraska Public Power District
Post Office Box 499
Columbus, Nebraska 68602-0499

Dear Mr. Horn:

SUBJECT: COOPER NUCLEAR STATION - AMENDMENT NO. 141 TO FACILITY OPERATING LICENSE NO. DPR-46 (TAC NO. 74842)

The Commission has issued the enclosed Amendment No. 141 to Facility Operating License No. DPR-46 for the Cooper Nuclear Station. The amendment consists of changes to the Technical Specifications in response to your application dated August 31, 1989, and amended by your letters of March 22, and April 19, 1991.

The amendment changes the Technical Specifications to revise the setpoint tolerance on the Low-Low Set Safety/Relief Valve pressure switches, revise the format and location of the specification, and correct typographical errors in instrument I.D. numbers.

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

Sincerely,
Original signed by

Paul W. O'Connor, Project Manager
Project Directorate IV-1
Division of Reactor Projects III, IV, and V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.141 to License No. DPR-46
2. Safety Evaluation

cc w/enclosures:
See next page

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OGC(MS15B18)	D. Hagan(MS3206)	G. Hill(4)(P1-37)	
Wanda Jones(MS7103)	J. Calvo(MS11F22)	PD4-1 Plant File	
GPA/PA(MS2G5)	ARM/LFMB(MS4503)	P. Harrell, RIV	

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OFC	PD4-1/LA	PD4-1/PE	PD4-1/PM	SPLB*	SICB*	OGC	PD4-1/D
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DATE	5/1/91	5/1/91	5/1/91	4/18/91	4/25/91	5/2/91	5/21/91

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

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Paul W. O'Connor

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Mr. Guy R. Horn
Nuclear Power Group Manager

cc:

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

NEBRASKA PUBLIC POWER DISTRICT

DOCKET NO. 50-298

COOPER NUCLEAR STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 141
License No. DPR-46

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

   A. The application for amendment by Nebraska Public Power District (the licensee) dated August 31, 1989, and amended by letters dated March 22, and April 19, 1991, 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 license 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-46 is hereby amended to read as follows:

   2. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 141 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Theodore R. Quay

Theodore R. Quay, Director
Project Directorate IV-1
Division of Reactor Projects III, IV, and V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: May 22, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 141

FACILITY OPERATING LICENSE NO. DPR-46

DOCKET NO. 50-298

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

REMOVE PAGES	INSERT PAGES
53	53
54	54
59	59
76	76
85	85
165	165
165a	165a
180	180

COOPER NUCLEAR STATION
TABLE 3.2.B (PAGE 1)
CIRCUITRY REQUIREMENTS CORE SPRAY SYSTEM

Instrument	Instrument I.D. No.	Setting Limit	Minimum Number of Operable Components Per Trip System	Action Required When Component Operability Is Not Assured (1)
Reactor Low Water Level	NBI-LIS-72 A, B, C, & D	≥-145.5 of Indicated Level	2	A
Reactor Low Pressure	NBI-PS-52 A2 & C2, NBI-PIS-52 B & D (Switch #2)	≤450 psig	2	A
Drywell High Pressure	PC-PS-101, A, B, C & D	≤2 psig	2	A
Core Spray Pump Disch. Pressure	CS-PS-44, A & B CS-PS-37, A & B	100 ≤ P ≤ 165 psig	2	A
Core Spray Pump Time Delay	CS-TDR-K16 A & B	9≤T≤11 seconds	1	B
Low Voltage Relay Emerg. Bus	27X1 - 1F & 1G 27X2 - 1F & 1G	Loss of Voltage	1	B
Aux. Bus Low Voltage Relay	27X3 - 1A & 1B	Loss of Voltage	1	B
Pump Discharge Line Low Pressure	CM-PS-73, A & B	≥10 psig	(3)	D

COOPER NUCLEAR STATION
TABLE 3.2.B (PAGE 2)
RESIDUAL HEAT REMOVAL SYSTEM (LPCI MODE) CIRCUITRY REQUIREMENTS

Instrument	Instrument I.D. No.	Setting Limit	Minimum Number of Operable Components Per Trip System (1)	Action Required When Component Operability Is Not Assured
Drywell High Pressure	PC-PS-101 A,B,C, & D	≤2 psig	2	A
Reactor Low Water Level	NBI-LIS-72, A,B,C, & D #1	≥-145.5" Indicated Level	2	A
Reactor Vessel Shroud Level Below Low Level Trip	NBI-LITS-73, A & B #1	≥-39 Indicated Level	1	B
Reactor Low Pressure	RR-PS-128, A & B	≤75 psig	1	B
Reactor Low Pressure (Injection Valve Permissive)	NBI-PS-52A2 & C2 NBI-PIS-52B & D (Switch #2)	≤450 psig	1	A
Drywell Pressure Containment Spray	PC-PS-119, A,B,C, &D	≤2 psig	2	A
RHR Pump Discharge	RHR-PS-120, A,B,C,&D RHR-PS-105, A,B,C,&D	100 ≤ P ≤ 165 psig 100 ≤ P ≤ 165 psig	2 2	A A
Reactor Low Pressure (Recirc. Discharge Permissive)	NBI-PS-52A1 & C1 NBI-PIS-52B & D (Switch #1)	185 ≤ P ≤ 235 psig	1	A

COOPER NUCLEAR STATION
TABLE 3.2.B (Page 7)
AUTOMATIC DEPRESSURIZATION SYSTEM (ADS) CIRCUITRY REQUIREMENTS

Instrument	Instrument I.D. No.	Setting Limit	Minimum Number of Operable Components Per Trip System (1)	Action Required When Component Operability Is Not Assured
Reactor Low Water Level	NBI-LIS-83, A & B	≥ +12.5" Indicated Level	1	B
	NBI-LIS-72, A,B,C & D	≥ -145.5" Indicated Level	2	A
ADS Timer	MS-TDR-K5, A & B	≤ 120 sec.	1	B

COOPER NUCLEAR STATION
TABLE 4.2.B (Page 7)
ADS SYSTEM AND LOW-LOW SET TEST & CALIBRATION FREQUENCIES

Item	Item I.D. No.	Functional Test Freq.	Instrument Calibration Freq.	Check
<u>Instruments</u>				
1. ADS Inhibit Switch	MS-SW-S3A & B	Once/Month (1)	N.A.	None
2. Reactor Low Water Level	NBI-LIS-83, A & B	Once/Month (1)	Once/3 Months	Once/Day
	NBI-LIS-72, A,B,C, & D	Once/Month (1)	Once/3 Months	Once/Day
3. ADS Timer	MS-TDR-K5 A & B	Once/Month (1)	Once/Oper. Cycle	None
4. Low-Low Set (LLS)	NBI-PS-51, A,B,C, & D	Once/Month (1)	Once/Oper. Cycle	None
<u>Logic (4)(6)</u>				
1. ADS Control Power Monitor		Once/6 Months	N.A.	
2. ADS Actuation		Once/6 Months	N.A.	
3. Low-Low Set Logic		Once/6 Months	N.A.	

B. Core and Containment Cooling Systems Initiation and Control

The instrumentation which initiates Core Standby Cooling System (CSCS) action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion, the Specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed. An exception to this is when logic functional testing is being performed.

CORE SPRAY - No Basis

LPCI MODE - No Basis

HPCI

The HPCI high flow and temperature instrumentation are provided to detect a break in the HPCI steam piping including the RHR Condensing Mode Steam. Tripping of this instrumentation results in actuation of HPCI isolation valves. Tripping logic for the high flow is a 1 out of 2 logic.

Temperature is monitored at twelve (12) locations with four (4) temperature sensors at each location. Two (2) sensors at each location are powered by "A" direct current control bus and two (2) by "B" direct current control bus. Each pair of sensors, e.g., "A" or "B", at each location are physically separated and the tripping of either "A" or "B" bus sensor will actuate HPCI isolation valves.

The trip settings of ≤ 300% of design flow for high flow and ≤ 200°F for high temperature are such that core uncovery is prevented and fission product release is within limits.

RCIC

The RCIC high flow and temperature instrumentation are arranged the same as that for the HPCI. The trip setting of ≤ 300% for high flow and ≤ 200°F for temperature are based on the same criteria as the HPCI.

ADS

The effective emergency core cooling for small pipe breaks, the HPCI system, must function since reactor pressure does not decrease rapid enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

LIMITING CONDITION FOR OPERATION

3.7.A (cont'd.)

6. Low-Low Set Relief Function

a. The low-low set function of the safety-relief valves shall be operable when there is irradiated fuel in the reactor vessel and the reactor coolant temperature is ≥ 212°F, except as specified in 3.7.A.6.a.1 and 2 below.

1. With the low-low function of one safety/relief valve (S/RV) inoperable, restore the inoperable LLS S/RV to OPERABLE within 14 days or be in the HOT STANDBY mode within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With the low-low set function of both S/RVs inoperable, be in at least HOT STANDBY within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. The pressure switches which control the low-low set safety/relief valves shall have the following settings.

NBI-PS-51A Open Low Valve
1015 ± 20 psig (Increasing)

NBI-PS-51B Close Low Valve
875 ± 20 psig (Decreasing)

NBI-PS-51C Open High Valve
1025 ± 20 psig (Increasing)

NBI-PS-51D Close High Valve
875 ± 20 psig (Decreasing)

B. Standby Gas Treatment System

1. Except as specified in 3.7.B.3 below, both standby gas treatment systems shall be operable at all times when secondary containment integrity is required.

2.a. The results of the in-place cold DOP leak tests on the HEPA filters shall show ≥99% DOP removal. The results of the halogenated hydrocarbon leak tests on the charcoal adsorbers shall show ≥99% halogenated hydrocarbon removal. The DOP and halogenated hydrocarbon tests shall be performed at a Standby Gas Treatment flowrate of ≤1780 CFM and at a Reactor Building pressure of ≤-.25" Wg.

SURVEILLANCE REQUIREMENT

4.7.A (cont'd.)

6. Low-Low Set Relief Function

a. The low-low set safety/relief valves shall be tested and calibrated as specified in Table 4.2.B.

B. Standby Gas Treatment System

1. At least once per operating cycle the following conditions shall be demonstrated.

a. Pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water at the system design flow rate.

b. Inlet heater input is capable of reducing R.H. from 100 to 70% R.H.

2.a. The tests and sample analysis of Specification 3.7.B.2 shall be performed at least once every 18 months for standby service or after every 720 hours of system operation and following significant painting, fire or chemical release in any ventilation zone communicating with the system.

3.7.B (cont'd)

b. The results of laboratory carbon sample analysis shall show ≥99% radioactive methyl iodide removal with inlet conditions of: velocity ≥42 FPM, ≥1.75 mg/m^3 inlet methyl iodide concentration, ≥70% R.H. and ≤30°C.

c. Each fan shall be shown to provide 1780 CMF ±10%.

3. From and after the date that one standby gas treatment system is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such system is sooner made operable, provided that during such seven days all active components of the other standby gas treatment system, and its associated diesel generator, shall be operable.

Fuel handling requirements are specified in Specification 3.10.E.

4. If these conditions cannot be met, procedures shall be initiated immediately to establish reactor conditions for which the standby gas treatment system is not required.

C. <u>Secondary Containment</u>

1. Secondary containment integrity shall be maintained during all modes of plant operation except when all of the following conditions are met.

4.7.B (cont'd)

b. Cold DOP testing shall be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

d. Each system shall be operated with the heaters on at least 10 hours every month.

e. Test sealing of gaskets for housing doors downstream of the HEPA filters and charcoal adsorbers shall be performed at, and in confromance with, each test performed for compliance with Specification 4.7.B.2.a and Specification 3.7.B.2.a.

3. System drains where present shall be inspected quarterly for adequate water level in loop-seals.

4.a. At least once per operating cycle automatic initiation of each branch of the standby gas treatment system shall be demonstrated.

b. At least once per operating cycle manual operability of the bypass valve for filter cooling shall be demonstrated.

c. When one standby gas treatment system becomes inoperable the other standby gas treatment system shall be demonstrated to be operable immediately and daily thereafter. A demonstration of diesel generator operability is not required by this specification.

C. <u>Secondary Containment</u>

1. Secondary containment surveillance shall be performed as indicated below:

3.7.A & 4.7.A BASES(cont'd)

The primary containment is normally slightly pressurized during periods of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase oxygen concentration. Once the containment is filled with nitrogen to the required concentration, no monitoring of oxygen concentration is necessary. However, at least twice a week the oxygen concentration will be determined as added assurance.

The 500 gallon conservative limit on the nitrogen storage tank assures that adequate time is available to get the tank refilled assuming normal plant operation. The estimated maximum makeup rate is 1500 SCFD which would require about 160 gallons for a 10 day makeup requirement. The normal leak rate should be about 200 SCFD.

3.7.A.6 & 4.7.A.6 LOW-LOW SET RELIEF FUNCTION

The low-low set relief logic is an automatic safety relief valve (SRV) control system designed to mitigate the postulated thrust load concern of subsequent actuations of SRV's during certain transients (such as inadvertant MSIV closure) and small and intermediate break loss-of-coolant accident (LOCA) events. The setpoints used in Section 3.7.A.6.b are based upon a minimum blowdown range to provide adequate time between valve actuations to allow the SRV discharge line high water leg to clear, coupled with consideration of instrument inaccuracy and the main steam isolation valve isolation setpoint.

The as-found setpoint for NBI-PS-51A, the pressure switch controlling the opening of RV-71D, must be ≤ 1040 psig. The as-found closing setpoint for NBI-PS-51B must be at least 90 psig less than 51A, and must be ≥ 850 psig. The as-found setpoint for NBI-PS-51C, pressure switch controlling the opening of RV-71F must be ≤ 1050 psig. The as-found closing setpoint for NBI-PS-51D must be at least 90 psig below 51C, and must be ≥ 850 psig. This ensures that the analytical upper limit for the opening setpoint (1050 psig), the analytical lower limit on the closing setpoint (850 psig) and the analytical limit on the blowdown range (≥ 90 psig) for the Low-Low Set Relief Function are not exceeded. Although the specified instrument setpoint tolerance is ± 20 psig, an instrument drift of ± 25 psig was used in the analysis to ensure adequate margin in determining the valve opening and closing setpoints. The opening setpoint is set such that, if both the lowest set non-LLS S/RV and the highest set of the two LLS S/RVs drift 25 psig in the worst case directions, the LLS S/RVs will still control subsequent S/RV actuations. Likewise, the closing setpoint is set to ensure the LLS S/RV closing setpoint remains above the MSIV low pressure trip. The 90 psig blowdown provides adequate energy release from the vessel to ensure time for the water leg to clear between subsequent S/RV actuations.

3.7.B & 3.7.C STANDBY GAS TREATMENT SYSTEM AND SECONDARY CONTAINMENT

The secondary containment is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation when the drywell is sealed and in service. The reactor building provides primary containment when the reactor is shut down and the drywell is open, as during refueling. Because the secondary containment is an integral part of the complete containment system. secondary containment is required at all times that primary containment is required as well as during refueling. Secondary containment may be broken for short periods of time to allow access to the reactor building roof to perform necessary inspections and maintenance.

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the stack during secondary containment isolation conditions. Both standby gas treatment system fans are designed to automatically start upon containment isolation and to maintain the reactor building pressure to the design negative pressure so that all leakage should be in-leakage. Should one system fail to start. the redundant system is designed to start automatically. Each of the two fans has 100 percent capacity.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 141 TO FACILITY OPERATING LICENSE NO. DPR-46

NEBRASKA PUBLIC POWER DISTRICT

COOPER NUCLEAR STATION

DOCKET NO. 50-298

1.0 INTRODUCTION

By letter dated August 31, 1989, as supplemented March 22, and April 19, 1991, Nebraska Public Power District (the licensee) requested an amendment to the Technical Specifications (TSs) appended to Facility Operating License No. DPR-46 for the Cooper Nuclear Station. The proposed amendment would revise the setpoint tolerance on the Low-Low Set of the Specification Safety/Relief Valve pressure switches, revise the format and location of the specification, and correct typographical errors in instrument I.D. numbers. The March 22, and April 19, 1991, letters provided clarification to the Bases section (page 180) and corrected an editorial error that did not change the initial proposed no significant hazards consideration determination.

2.0 DISCUSSION

The low-low set (LLS) relief logic is an automatic safety relief valve (SRV) control system designed to mitigate the postulated thrust load concern of repeated actuations of SRV's during certain transients. Reactor pressure indications in excess of the reactor scram setpoint (1045 psig) and indications of any SRV opening (30 psig in discharge line) arm the LLS logic. The LLS logic controls the pneumatic actuators of two SRVs to control reactor pressure and reduce the subsequent openings of SRVs and thereby limit the forces applied to the discharge piping and torus. The LLS performs this function by introducing an extended blowdown from the two LLS SRVs by utilizing reduced relief and reseat setpoints. The energy released during the period the LLS SRVs are open will delay the repressurization of the reactor coolant system and prevent subsequent SRV actuation.

Existing Technical Specification 3.2 (Table 3.2.B) defines the required LLS setpoint and tolerance values. The LLS function relief setpoints are 1015 psig and 1025 psig respectively for relief valves 71D and 71F. The closing setpoint for both valves is 875 psig and the acceptable tolerance for opening and closing setpoints is specified to be ±10 psi.

The licensee has proposed to relocate the setpoints from Table 3.2.B to a specific LLS TS (3.7.A.6/4.7.A.6). In addition, the title of Table 4.2.B is changed to "ADS System/Low-Low Set Test and..." to better reflect the inclusion of the LLS function in the surveillance requirement. The proposed revision includes changes to Table 3.2.B to correct and/or clarify instruments. The staff concurs with the licensee that the above editorial changes are appropriate.

The proposal revises the allowable setpoint tolerances associated with the LLS pressure switches from ±10 psi to ±20 psi. The setpoints and the associated tolerances are intended to ensure that the LLS function prevents opening of other SRVs following the initial pressure increase. This is accomplished by establishing the open and close LLS setpoints below the other SRV setpoints and sufficiently apart from each other (>90 psi) to remove enough energy from the RCS to delay repressurization and thereby provide adequate time between valve actuations to allow the SRV discharge line high water leg to clear. Another consideration is to determine the LLS reseat setpoint such that an unnecessary isolation of the Main Steam Isolation Valves (MSIVs) does not occur. The licensee states that the above requirements can be satisfied while allowing a tolerance of ±20 psi which is more realistic in terms of the accuracy limitations of available pressure switches. The proposed change involves only the allowable tolerance for the LLS pressure switches and does not revise the actual LLS setpoints or the initial lifting setpoints (safety-relief function) of the two valves associated with the LLS function.

In order to justify the change in the allowable tolerance, the licensee demonstrated that the above design considerations continue to be satisfied with a LLS setpoint tolerance of ±20 psi. Margin remains between the maximum LLS setpoint (1025 psig) assuming the highest proposed allowable deviation due to accuracy limitations and drift (20 psi) and the minimum SRV setpoint (1080 psig) with consideration for the TS allowable SRV setpoint deviation (±1%). A greater than 90 psi blowdown range is also maintained even assuming a maximum allowable negative deviation of 20 psi on the minimum LLS opening setpoint (1015 psig) and a maximum allowable positive deviation of 20 psi on the LLS closing setpoint (875 psig). The desired margin between LLS closing and MSIV isolation is also maintained assuming maximum deviations of ±20 psi of both LLS and MSIV isolation related setpoints. Based upon its review, the staff finds the proposed change in allowable LLS setpoint tolerances does not prevent the system from performing its function and is therefore acceptable.

3.0 STATE CONSULATION

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

4.0 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 in 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 (54 FR 49134). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Section 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 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: W. Reckley

Date: May 22, 1991