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

June 3, 1996

Dr. Robert C. Mecredy
Vice President, Nuclear Operations
Rochester Gas and Electric Corporation
89 East Avenue
Rochester, NY 14649

SUBJECT: ISSUANCE OF AMENDMENT NO. 65 TO FACILITY OPERATING LICENSE NO. DPR-18, R. E. GINNA NUCLEAR POWER PLANT (TAC NO. M95346)

Dear Dr. Mecredy:

The Commission has issued the enclosed Amendment No. 65 to Facility Operating License No. DPR-18 for the R. E. Ginna Nuclear Power Plant. This amendment is in response to your application dated May 8, 1996, as supplemented May 10, 1996, May 29, 1996, and June 3, 1996.

This amendment modifies the Technical Specifications to correct several typographical errors that were implemented in the Improved Technical Specifications at Ginna Station per Amendment No. 61.

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

Sincerely,

Guy S. Vissing

Guy S. Vissing, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosures: 1. Amendment No. 65 to License No. DPR-18
2. Safety Evaluation

cc w/encls: See next page

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PDR ADOCK 05000244
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ORIGINAL SIGNED BY:

Guy S. Vissing, Senior Project Manager
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Guy S. Vissing, Senior Project Manager
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Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-244

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with 1 change in SE

Dr. Robert C. Mecredy
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Rochester Gas and Electric Corporation
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This amendment modifies the Technical Specifications to correct several typographical errors that were implemented in the Improved Technical Specifications at Ginna Station per Amendment No. 61.

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

Sincerely,

Guy S. Vissing, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosures: 1. Amendment No. to License No. DPR-18
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SE does not clearly justify admin errors, but inappropriately describes "obvious" errors 3.3.1 (ii)(c) f(ΔI) is wrong

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DATED: June 3, 1996

AMENDMENT NO. 65 TO FACILITY OPERATING LICENSE NO. DPR-18-GINNA NUCLEAR POWER PLANT

Docket File
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C. Grimes, 11/E/22
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cc: Plant Service list

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

ROCHESTER GAS AND ELECTRIC CORPORATION

DOCKET NO. 50-244

R. E. GINNA NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 65
License No. DPR-18

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

   A. The application for amendment filed by the Rochester Gas and Electric Corporation (the licensee) dated May 8, 1996, as supplemented May 10, 1996, May 29, 1996, and June 3, 1996, 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-18 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 65, 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 its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Jocelyn A. Mitchell

Jocelyn A. Mitchell, Acting 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: June 3, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 65

FACILITY OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

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
i	i
ii	ii
iii	iii
3.2-13	3.2-13
3.3-7	3.3-7
3.3-18	3.3-18
3.3-19	3.3-19
3.3-24	3.3-24
3.3-32	3.3-32
3.3-33	3.3-33
3.3-40	3.3-40
B 3.3-100	B 3.3-100

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(continued)

TABLE OF CONTENTS

(continued)

TABLE OF CONTENTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.6 --------NOTES-------- 1. Only required to be performed if the cause of the QPTR alarm is not associated with inoperable QPTR instrumentation. 2. Required Action A.6 must be completed when Required Action A.5 is completed and Note 1, above, does not apply. 3. Only one of the Completion Times, whichever becomes applicable first, must be met. -------------------- Perform SR 3.2.1.1 and SR 3.2.2.1.	Within 24 hours after reaching RTP <u>OR</u> Within 48 hours after increasing THERMAL POWER above the limits of Required Actions A.1 and A.2

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Q. Required Action and Associated Completion Time of Condition P not met.	Q.1 Reduce THERMAL POWER to < 50% RTP.	6 hours
	<u>AND</u>	
	Q.2.1 Verify Steam Dump System is OPERABLE.	7 hours
	<u>OR</u>	
	Q.2.2 Reduce THERMAL POWER to < 8% RTP.	7 hours
R. As required by Required Action A.1 and referenced by Table 3.3.1-1.	R.1 --------NOTE--------- One train may be bypassed for up to 4 hours for surveillance testing provided the other train is OPERABLE. --------------------- Restore train to OPERABLE status.	6 hours
S. As required by Required Action A.1 and referenced by Table 3.3.1-1.	S.1 Verify interlock is in required state for existing plant conditions.	1 hour
	<u>OR</u>	
	S.2 Declare associated RTS Function channel(s) inoperable.	1 hour

(continued)

Table 3.3.1-1 (page 5 of 6)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Trip Setpoint is defined by:

$$\text{Overtemperature } \Delta T \leq \Delta T_0 \left\{ K_1 + K_2 (P - P') - K_3 (T - T') \left[\frac{1 + \tau_1 s}{1 + \tau_2 s}\right] - f(\Delta I) \right\}$$

Where:

ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.

s is the Laplace transform operator, sec^{-1}.

T is the measured RCS average temperature, °F.
T′ is the nominal T_{avg} at RTP, °F.

P is the measured pressurizer pressure, psig.
P′ is the nominal RCS operating pressure, psig.

K_1 is the Overtemperature ΔT reactor trip setpoint, 1.20.
K_2 is the Overtemperature ΔT reactor trip depressurization setpoint penalty coefficient, 0.000900/psi.
K_3 is the Overtemperature ΔT reactor trip heatup setpoint penalty coefficient, 0.0209/°F.

τ_1 is the measured lead/lag time constant, 25 seconds.

τ_2 is the measured lead/lag time constant, 5 seconds.

f(ΔI) is a function of the indicated difference between the top and bottom detectors of the Power Range Neutron Flux channels where q_t and q_b are the percent power in the top and bottom halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

$f(\Delta I) = 0$	when $q_t - q_b$ is $\leq$ +13% RTP
$f(\Delta I) = 1.3 \{(q_t - q_b) - 13\}$	when $q_t - q_b$ is > +13% RTP

Table 3.3.1-1 (page 6 of 6)
Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Trip Setpoint is defined by:

$$\text{Overpower } \Delta T \le \Delta T_0 \left\{ K_4 - K_5 (T - T') - K_6 \left[\frac{\tau_3 s T}{\tau_3 s + 1} \right] - f(\Delta I) \right\}$$

Where:

ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.

s is the Laplace transform operator, sec^{-1}.

T is the measured RCS average temperature, °F.
T' is the nominal T_{avg} at RTP, °F.

K_4 is the Overpower ΔT reactor trip setpoint, 1.077.
K_5 is the Overpower ΔT reactor trip heatup setpoint penalty coefficient which is:
0.0/°F for $T < T'$ and;
0.0011/°F for $T \ge T'$.
K_6 is the Overpower ΔT reactor trip thermal time delay setpoint penalty which is:
0.0262/°F for increasing T and;
0.00/°F for decreasing T.

τ_3 is the measured lead/lag time constant, 10 seconds.

$f(\Delta I)$ is a function of the indicated difference between the top and bottom detectors of the Power Range Neutron Flux channels where q_t and q_b are the percent power in the top and bottom halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

$f(\Delta I) = 0$	when $q_t - q_b$ is $\le +13\%$ RTP
$f(\Delta I) = 1.3 \{(q_t - q_b) - 13\}$	when $q_t - q_b$ is $> +13\%$ RTP

SURVEILLANCE REQUIREMENTS

------------------------------------NOTE------------------------------------
Refer to Table 3.3.2-1 to determine which SRs apply for each ESFAS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.2.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.2.2	Perform COT.	92 days
SR 3.3.2.3	------------------NOTE------------------ Verification of relay setpoints not required. --- Perform TADOT.	92 days
SR 3.3.2.4	------------------NOTE------------------ Verification of relay setpoints not required. --- Perform TADOT.	24 months
SR 3.3.2.5	Perform CHANNEL CALIBRATION.	24 months
SR 3.3.2.6	Verify the Pressurizer Pressure – Low and Steam Line Pressure – Low Functions are not bypassed when pressurizer pressure > 2000 psig.	24 months
SR 3.3.2.7	Perform ACTUATION LOGIC TEST.	24 months

Table 3.3.3-1 (page 1 of 2)
Post Accident Monitoring Instrumentation

	FUNCTION	REQUIRED CHANNELS	CONDITION
1.	Pressurizer Pressure	2	G
2.	Pressurizer Level	2	G
3.	Reactor Coolant System (RCS) Hot Leg Temperature	1 per loop	G
4.	RCS Cold Leg Temperature	1 per loop	G
5.	RCS Pressure (Wide Range)	2	G
6.	RCS Subcooling Monitor	2	G
7.	Reactor Vessel Water Level	2	H
8.	Containment Sump B Water Level	2	G
9.	Containment Pressure (Wide Range)	2	G
10.	Containment Area Radiation (High Range)	2	H
11.	Hydrogen Monitors	2	G
12.	Condensate Storage Tank Level	2	G
13.	Refueling Water Storage Tank Level	2	G
14.	Residual Heat Removal Flow	2	G
15.	Core Exit Temperature – Quadrant 1	2[a]	G
16.	Core Exit Temperature – Quadrant 2	2[a]	G
17.	Core Exit Temperature – Quadrant 3	2[a]	G
18.	Core Exit Temperature – Quadrant 4	2[a]	G
19.	Auxiliary Feedwater (AFW) Flow to Steam Generator (SG) A	2	G
20.	AFW Flow to SG B	2	G
21.	SG A Water Level (Narrow Range)	2	G
22.	SG B Water Level (Narrow Range)	2	G

(continued)

(a) A channel consists of two core exit thermocouples (CETs).

Table 3.3.3-1 (page 2 of 2)
Post Accident Monitoring Instrumentation

FUNCTION	REQUIRED CHANNELS	CONDITION
23. SG A Water Level (Wide Range)	2	G
24. SG B Water Level (Wide Range)	2	G
25. SG A Pressure	2	G
26. SG B Pressure	2	G

Table 3.3.5-1 (page 1 of 1)
Containment Ventilation Isolation Instrumentation

FUNCTION	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	TRIP SETPOINT
1. Automatic Actuation Logic and Actuation Relays	2 trains	SR 3.3.5.3	NA
2. Containment Radiation			
a. Gaseous	1	SR 3.3.5.1 SR 3.3.5.2 SR 3.3.5.4	(a)
b. Particulate	1	SR 3.3.5.1 SR 3.3.5.2 SR 3.3.5.4	(a)
3. Containment Isolation	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 3, for all initiation functions and requirements.		
4. Containment Spray - Manual Initiation	Refer to LCO 3.3.2, "ESFAS Instrumentation," Function 2.a, for all initiation functions and requirements.		

Notes:

(a) Per Radiological Effluent Controls Program.

BASES

ACTIONS (continued)

M.1

If the Required Actions and Completion Times of Condition L are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and pressurizer pressure reduced to < 2000 psig within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

N.1

Condition N applies if a AFW Manual Initiation channel is inoperable. If a manual initiation switch is inoperable, the associated AFW or SAFW pump must be declared inoperable and the applicable Conditions of LCO 3.7.5, "Auxiliary Feedwater (AFW) System" must be entered immediately. Each AFW manual initiation switch controls one AFW or SAFW pump. Declaring the associated pump inoperable ensures that appropriate action is taken in LCO 3.7.5 based on the number and type of pumps involved.

SURVEILLANCE REQUIREMENTS

The SRs for each ESFAS Function are identified by the SRs column of Table 3.3.2-1. Each channel of process protection supplies both trains of the ESFAS. When testing Channel 1, Train A and Train B must be examined. Similarly, Train A and Train B must be examined when testing Channel 2, Channel 3, and Channel 4 (if applicable). The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

A Note has been added to the SR Table to clarify that Table 3.3.2-1 determines which SRs apply to which ESFAS Functions.

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 65 TO FACILITY OPERATING LICENSE NO. DPR-18

ROCHESTER GAS AND ELECTRIC CORPORATION

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

1.0 INTRODUCTION

By letter dated May 8, 1996, as supplemented May 10, 1996, May 29, 1996, and June 3, 1996, the Rochester Gas and Electric Corporation (the licensee or RG&E) submitted a request for changes to the R. E. Ginna Nuclear Power Plant Technical Specifications (TSs). The requested changes would modify the TS to correct several typographical errors that were implemented in the Improved Technical Specifications (ITS) at Ginna Station per Amendment No 61. The May 10, 1996, May 29, 1996 and June 3, 1996, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

The licensee requested changes to 3 pages of the Table of Contents, 8 other changes to the TS and 1 change to the Bases. The proposed changes, along with the staff's evaluation follows:

1. Table of Contents

   i. The title for Limiting Conditions for Operation (LCOs) 3.3.5 and 3.7.3 would be revised to provide consistency with the actual title found within the TS.

   ii. The page number for LCOs 3.6.4, 3.6.5, 3.6.6, 3.6.7, 3.7.8, 3.7.9, 3.7.10, 3.7.11, 3.7.12, 3.7.13, and 3.7.14 would be revised to provide consistency with the actual page these LCOs are found within the TS.

   Since these proposed changes provide consistency between the Table of Contents and the actual sections, the staff has found them to be acceptable.

2. LCO 3.2.4

   i. The extra word "increased" in the Completion Time column for Required Action A.6 would be deleted. This word was inadvertently added and was not consistent with that recommended by NUREG-1431, Standard Technical Specifications, Westinghouse Plants, April 1995.

Since the proposed change of removing "increased" would provide consistency between NUREG-1431 and the TS and would not change the requirement, the staff has determined that the change is acceptable.

3. LCO 3.3.1

   i. The Note for Required Action R.1 would be revised to provide consistency with NUREG-1431 and the current bases and would more clearly define the required action. Since the change would clarify a point of confusion, the staff has determined the proposed change to be acceptable.

   ii. Several errors in the f(ΔI) input into the Overtemperature ΔT and Overpower ΔT equations for Table 3.3.1-1 would be corrected. These include:

      a. The "Q_b" in the f(ΔI) discussion for Overtemperature ΔT would be changed to "q_b" based on the equations presented below this text.

      b. The description of when to set f(ΔI) = 0 would be changed to "when $q_t - q_b$ is ≤ +13% RTP" versus when "> +13% RTP." The error with respect to the ≤ sign is readily apparent since the same "when $q_t - q_b$ is > +13% RTP" text is used in the next line for when f(ΔI) must be shifted. The correction of this error is based on the "old" TS, page 2.3-2.

      c. The values for certain constants were revised to provide necessary units to ensure that the equation is mathematically correct and consistent with all defined values.

   The errors identified above were the results of incorrectly translating the requirements of the "old" TS to the current TS. The proposed changes were reviewed by the staff and since they would retain the technical operating requirements as required by the "old" TS, the staff has found the changes acceptable.

4. LCO 3.3.2

   i. The Note prior to the Surveillance Requirements in LCO 3.3.2 would be revised to remove the plural application of "note" and to delete the "1" preceding the note consistent with the NUREG-1431 format for single notes. This change would also affect a change in the bases (Page B 3.3-100). Since this proposed change provides clarity, the staff has determined that the proposed change is acceptable.

5. LCO 3.3.3

   i. The titles of all steam generator (SG) related instrumentation listed in Table 3.3.3-1 would be changed since the current wording implies that these are instrumentation to a given SG. As such, the text would be revised to reflect that the instrumentation is from a given SG. Since the proposed change more correctly characterizes the instrumentation, the staff has found the proposed changes to be acceptable.

6. LCO 3.3.5

   i. The title for Table 3.3.5-1 Function 4 would be changed to "Containment Spray - Manual Initiation." The current wording is "Manual Isolation" which is not correct per Table 3.3.2-1, Function 2.a and the bases for LCO 3.3.5. Since this change would provide consistency between Table 3.3.5-1 and Table 3.3.2-1 and corrects an error in the proper identification of the operation of Containment Spray, the staff has determined that the proposed change is acceptable.

3.0 EXIGENT CIRCUMSTANCES

Pursuant to 10 CFR 50.91(a)(6), the licensee requested that the proposed amendment is on an exigent basis. The proposed change would permit the R. E. Ginna Nuclear Power Plant to support a planned entry into MODE 2 on June 4, 1996. The condition that lead to discovery of the need was the result of recent identification by various plant staff personnel of several typographical errors and two errors relating to equations for Overtemperature ΔT and Overpower ΔT within the ITS issued on February 24, 1996, by Amendment No. 61. The latter errors would provide confusion to the operators by creating an equation in which a specific input is not defined.

Based on the above, the NRC staff has determined that the licensee has used best efforts to make a timely application and that exigent circumstances are present which warrant processing the requested amendment pursuant to 10 CFR 50.91(a)(6).

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission has made a final determination that the amendment involves no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The Commission has evaluated the proposed changes against the above standards as required by 10 CFR 50.91(a) and has concluded that the changes do not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

   Operation of Ginna Station in accordance with the proposed changes does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes only correct various typographical errors within the technical specifications. The errors were discovered during use of the new improved technical specifications and do not involve any technical issues when compared to NUREG-1431 or the "old" technical specifications. As such, these changes are administrative and do not impact initiators or analyzed events or assumed mitigation of accident or transient events. Therefore, these changes to not involve a significant increase in the probability or consequences of an accident previously analyzed.

2. Operation of Ginna Station in accordance with the proposed changes does not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed changes to not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. The proposed changes will not impose any new or different requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of Ginna Station in accordance with the proposed changes does not involve a significant reduction in a margin of safety. The proposed changes will not reduce a margin of plant safety because the changes are administrative in nature. As such, no question of safety is involved, and the change does not involve a significant reduction in a margin of safety.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York 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 made a final finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (61 FR 25966). 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.

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