Docket No. 50-382 August 8, 1991

Mr. Ross P. Barkhurst
Vice President Operations
Entergy Operations, Inc.
Post Office Box B
Killona, Louisiana 70066

Dear Mr. Barkhurst:

SUBJECT: ISSUANCE OF AMENDMENT NO. 69 TO FACILITY OPERATING LICENSE NPF-38 - WATERFORD STEAM ELECTRIC STATION, UNIT 3 (TAC NO. 75190)

The Commission has issued the enclosed Amendment No. 69 to Facility Operating License No. NPF-38 for the Waterford Steam Electric Station, Unit 3. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated October 16, 1989, as supplemented by letter dated September 14, 1990.

The amendment changes the Appendix A Technical Specifications by changing the frequency of select channel functional tests from monthly to quarterly on the Reactor Protection System and Engineered Safety Feature Actuation Systems.

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
David L. Wigginton, Project Manager
Project Directorate IV-1
Division of Reactor Projects III, IV, and V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 69 to NPF-38
2. Safety Evaluation

cc w/enclosures:
See next page

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

August 8, 1991

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Mr. Ross P. Barkhurst
Vice President Operations
Entergy Operations, Inc.
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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,

Wayne C Walker for

David L. Wigginton, Project Manager
Project Directorate IV-1
Division of Reactor Projects III, IV, and V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 69 to NPF-38
2. Safety Evaluation

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

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-382

WATERFORD STEAM ELECTRIC STATION, UNIT 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.69
License No. NPF-38

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

   A. The application for amendment by Entergy Operations, Inc. (the licensee) dated October 16, 1989, as supplemented September 4, 1990, 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. NPF-38 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 69 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Paul W. O'Connor

for 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: August 8, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 69

TO FACILITY OPERATING LICENSE NO. NPF-38

DOCKET NO. 50-382

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 areas of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE PAGES	INSERT PAGES
3/4 3-10	3/4 3-10
3/4 3-11	3/4 3-11
3/4 3-12	3/4 3-12
3/4 3-25	3/4 3-25
3/4 3-26	3/4 3-26
3/4 3-27	3/4 3-27
B 3/4 3-1	B 3/4 3-1

TABLE 3.3-2 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
11. Steam Generator Level - High	Not Applicable
12. Reactor Protection System Logic	Not Applicable
13. Reactor Trip Breakers	Not Applicable
14. Core Protection Calculators	Not Applicable
15. CEA Calculators	Not Applicable
16. Reactor Coolant Flow - Low	0.70 second

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

**Response time shall be measured from the time the CPC/CEAC receives an input signal until the electrical power is interrupted to the CEA drive mechanism.

#Response time shall be measured from the output of the sensor. RTD response time for all the RTDs shall be measured at least once per 18 months. The measured P_τ of the slowest RTD shall be less than or equal to 8 seconds (P_τ assumed in the safety analysis).

##Response time shall be measured from the output of the pressure transmitter. The transmitter response time shall be less than or equal to 0.70 second.

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	R and S/U(1)	1, 2, 3*, 4*, 5*
2. Linear Power Level - High	S	D(2,4),M(3,4), Q(4)	Q	1, 2
3. Logarithmic Power Level - High	S	R(4)	Q and S/U(1)	2#, 3, 4, 5
4. Pressurizer Pressure - High	S	R	Q	1, 2
5. Pressurizer Pressure - Low	S	R	Q	1, 2
6. Containment Pressure - High	S	R	Q	1, 2
7. Steam Generator Pressure - Low	S	R	Q	1, 2
8. Steam Generator Level - Low	S	R	Q	1, 2
9. Local Power Density - High	S	D(2,4), R(4,5)	Q, R(6)	1, 2
10. DNBR - Low	S	S(7), D(2,4), M(8), R(4,5)	Q, R(6)	1, 2
11. Steam Generator Level - High	S	R	Q	1, 2
12. Reactor Protection System Logic	N.A.	N.A.	Q and S/U(1)	1, 2, 3*, 4*, 5*

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
13. Reactor Trip Breakers	N.A.	N.A.	M(10), S/U(1)	1, 2, 3*, 4*, 5*
14. Core Protection Calculators	S	D(2,4),R(4,5)	Q(9),R(6)	1, 2
15. CEA Calculators	S	R	Q,R(6)	1, 2
16. Reactor Coolant Flow - Low	S	R	Q	1, 2

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

*With the reactor trip breakers in the closed position, the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

#The provisions of Specification 4.0.4 are not applicable when reducing reactor power to less than 10^{-4}% of RATED THERMAL POWER from a reactor power level greater than 10^{-4}% of RATED THERMAL POWER. Upon reducing power below 10^{-4}% of RATED THERMAL POWER, a CHANNEL FUNCTIONAL TEST shall be performed within 2 hours if not performed during the previous 31 days. This requirement does not apply with the reactor trip breakers open.

(1) Each startup or when required with the reactor trip breakers closed and the CEA drive system capable of rod withdrawal, if not performed in the previous 7 days.

(2) Heat balance only (CHANNEL FUNCTIONAL TEST not included), above 15% of RATED THERMAL POWER: adjust the Linear Power Level signals and the CPC addressable constant multipliers to make the CPC ΔT power and CPC nuclear power calculations agree with the calorimetric calculation if absolute difference is greater than 2%. During PHYSICS TESTS, these daily calibrations may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) Above 15% of RATED THERMAL POWER, verify that the linear power subchannel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the Core Protection Calculators.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) After each fuel loading and prior to exceeding 70% of RATED THERMAL POWER, the incore detectors shall be used to determine the shape annealing matrix elements and the Core Protection Calculators shall use these elements.

(6) This CHANNEL FUNCTIONAL TEST shall include the injection of simulated process signals into the channel as close to the sensors as practicable to verify OPERABILITY including alarm and/or trip functions.

(7) Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation or by calorimetric calculations and if necessary, adjust the CPC addressable constant flow coefficients such that each CPC indicated flow is less than or equal to the actual flow rate. The flow measurement uncertainty is included in the BERR1 term in the CPC and is equal to or greater than 4%.

(8) Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by calorimetric calculations.

(9) The quarterly CHANNEL FUNCTIONAL TEST shall include verification that the correct values of addressable constants are installed in each OPERABLE CPC.

(10) At least once per 18 months and following maintenance or adjustment of the reactor trip breakers, the CHANNEL FUNCTIONAL TEST shall include independent verification of the undervoltage trip function and the shunt trip function.

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTAION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. SAFETY INJECTION (SIAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	1, 2, 3, 4
b. Containment Pressure - High	S	R	Q	1, 2, 3
c. Pressurizer Pressure - Low	S	R	Q	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(2) (3) (6)	1, 2, 3
2. CONTAINMENT SPRAY (CSAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	1, 2, 3, 4
b. Containment Pressure -- High - High	S	R	Q	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	M(1) (2) (3)	1, 2, 3
3. CONTAINMENT ISOLATION (CIAS)				
a. Manual CIAS (Trip Buttons)	N.A.	N.A.	R	1, 2, 3, 4
b. Containment Pressure - High	S	R	Q	1, 2, 3
c. Pressurizer Pressure - Low	S	R	Q	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1) (2) (3)	1, 2, 3
4. MAIN STEAM LINE ISOLATION				
a. Manual (Trip Buttons)	N.A.	N.A.	R	1, 2, 3
b. Steam Generator Pressure - Low	S	R	Q	1, 2, 3
c. Containment Pressure - High	S	R	Q	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1) (2) (3)	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
5. SAFETY INJECTION SYSTEM RECIRCULATION (RAS)				
a. Manual RAS (Trip Buttons)	N.A.	N.A.	R	1, 2, 3, 4
b. Refueling Water Storage Pool - Low	S	R	Q	1, 2, 3, 4
c. Automatic Actuation Logic	N.A.	N.A.	M(1) (2) (3)	1, 2, 3, 4
6. LOSS OF POWER (LOV)				
a. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	N.A.	R	D(4)	1, 2, 3
b. 480 V Emergency Bus Undervoltage (Loss of Voltage)	N.A.	R	D(4)	1, 2, 3
c. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	N.A.	R	D(4)	1, 2, 3

TABLE 4.3.-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
7. EMERGENCY FEEDWATER (EFAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	1, 2, 3
b. SG Level (1/2)-Low and ΔP (1/2) - High	S	R	Q	1, 2, 3
c. SG Level (1/2) - Low and No Pressure - Low Trip (1/2)	S	R	Q	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1) (2) (3)	1, 2, 3
e. Control Valve Logic (Wide Range SG Level - Low)	S	R	SA(5)	1, 2, 3

TABLE NOTATION

(1) Each train or logic channel shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(2) Testing of Automatic Actuation Logic shall include energization/deenergization of each initiation relay and verification of the OPERABILITY of each initiation relay.

(3) A subgroup relay test shall be performed which shall include the energization/deenergization of each subgroup relay and verification of the OPERABILITY of each subgroup relay. Relays K109, K114, K202, K301, K305, K308 and K313 are exempt from testing during power operation but shall be tested at least once per 18 months and during each COLD SHUTDOWN condition unless tested within the previous 62 days.

(4) Using installed test switches.

(5) To be performed during each COLD SHUTDOWN if not performed in the previous 6 months.

(6) Each train shall be tested, with the exemption of relays, K110, K410 and K412, at least every 62 days on a STAGGERED TEST BASIS. Relays K110, K410 and K412 shall be tested at least every 62 days but will be exempt from the STAGGERED TEST BASIS.

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-3.

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEMS INSTRUMENTATION

The OPERABILITY of the Reactor Protective and Engineered Safety Features Actuation Systems instrumentation and bypasses ensures that (1) the associated Engineered Safety Features Actuation action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses.

The redundancy design of the Control Element Assembly Calculators (CEAC) provides reactor protection in the event one or both CEACs become inoperable. If one CEAC is in test or inoperable, verification of CEA position is performed at least every 4 hours. If the second CEAC fails, the CPCs will use DNBR and LPD penalty factors to restrict reactor operation to some maximum fraction of RATED THERMAL POWER. If this maximum fraction is exceeded, a reactor trip will occur.

The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. The quarterly frequency for the channel functional tests for these systems comes from the analyses presented in topical report CEN-327: RPS/ESFAS Extended Test Interval Evaluation, as supplemented.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping, or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) in place, onsite, or offsite test measurements or (2) utilizing replacement sensors with certified response times.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that: (1) the radiation levels are continually measured in the areas served by the individual channels; (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," December 1980 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.3.2 INCORE DETECTORS

The OPERABILITY of the incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix "A" of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974.

3/4.3.3.4. METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50.

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. NPF-38

ENTERGY OPERATIONS, INC.

WATERFORD STEAM ELECTRIC STATION, UNIT 3

DOCKET NO. 50-382

1.0 INTRODUCTION

By application dated October 16, 1989, Louisiana Power and Light Company submitted a request for changes to the Waterford Steam Electric Station, Unit 3, Technical Specification (TS). The requested changes would revise the channel functional tests from monthly to quarterly for the Reactor Protection System (RPS) except for the reactor trip breakers, and for the Engineering Safety Features Actuation Systems (ESFAS) except for the automatic actuation logic tests. The exceptions remain unchanged. By letter dated September 14, 1990, Entergy Operations, Inc. (EOI and now the licensee) provided assurance that the setpoint drift suffered by any of the instrument channels addressed in CEN-327 over the extended test interval should not exceed the allowable value as calculated by the setpoint methodology. This September 14, 1990, letter provides the related assurance needed for acceptability of the CEN-327 approval and does not change in any way the notice of no significant hazards published in the Federal Register on February 21, 1990 (55 FR 6106).

2.0 EVALUATION

By letter to the CE Owners Group (CEOG) dated November 6, 1989, the NRC approved the CEOG's Topical Report CEN-327, "RPS/ESFAS Extended Test Interval Evaluation". The Safety Evaluation transmitted by that letter found that extending the surveillance test interval for the RPS and the ESFAS channels was acceptable for CE plants and therefore, for Waterford Steam Electric Station, Unit No. 3. In the case of Waterford, however, the licensee did not obtain the necessary support for the monthly automatic actuation tests and the reactor trip breakers tests and these tests remain unchanged.

In our review of CEN-327, it was determined that each CE licensee to which CEN-327 applies must also confirm that the instrument drift occurring over the proposed test interval would not cause the setpoint values to exceed those assumed in the safety analysis and specified in the Technical Specifications. The licensees were to confirm their review of the drift against the allowable value as calculated for that channel by their setpoint methodology. The Waterford 3 application was received before the CEOG Owners Group letter was issued and this confirmation was not included. The licensee, by letter dated

September 14, 1990, provided that confirmation. We have reviewed the licensee's proposed changes to the Technical Specifications and their confirmation of the setpoint drift issue and conclude that the proposed changes are acceptable.

3.0 STATE CONSULTATION

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

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 (55 FR 6106). 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.

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: D. Wigginton

Date: August 8, 1991