February 29, 1988

Docket No. 50-316

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Mr. Milton P. Alexich, Vice President
Indiana Michigan Power Company
c/o American Electric Power Service Corporation
1 Riverside Plaza
Columbus, Ohio 43216

Dear Mr. Alexich:

SUBJECT: ISSUANCE OF AMENDMENT NO. 99 TO FACIITY OPERATING LICENSE NO. DPR-74: DELAY IN CERTAIN 18-MONTH SURVEILLANCES (TAC NO. 67039)

The Commission has issued the enclosed Amendment No. 99 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 dated January 11, 1988.

This amendment changes the Technical Specifications to allow certain tests normally designated as 18-month surveillances to be delayed until the end of the next refueling outage currently scheduled to begin during the second quarter of 1988. The Commission has determined that failure to act in a timely manner to grant extensions for the RHR auto-closure interlock testing, steam generator snubber functional testing and rod position indication functional testing would result in shutdown of Unit 2. Therefore this amendment deals only with these extensions. The remaining changes requested by your application of January 11, 1988, will be the subject of future Commission action.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance and Final Determination of No Significant Hazards Consideration and Opportunity for Hearing will be included in the Commission's next biweekly Federal Register notice.

Sincerely,
Original signed by

John F. Stang, Project Manager
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

Enclosures:
1. Amendment No. 99 to DPR-74
2. Safety Evaluation

8803040359 880229
PDR ADOCK 05000316
P PDR

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See next page
See Previous Concurrence*

LA/PD31:DRSP*	PM/PD31:DRSP*	MEB*	SICB*	SRXB*
RIngram	JStang:lt	TMarsh	SNewberry	MWHodges
2/19/88	2/19/88	2/23/88	02/19/88	2/23/88

D/PD31:DRSP*	AD:DRSP*	OGC*
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2/26/88	2/26/88	2/24/88

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See next page

See changes p. 4 of SER

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MVirgilio	GHolahan			
2/26/88	2/26/88	2/24/88		

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Mr. Milton Alexich
Indiana Michigan Power Company

Donald C. Cook Nuclear Plant

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Department of Public Health
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 99
License No. DPR-74

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

   A. The application for amendment by Indiana Michigan Power Company (the licensee) dated January 11, 1988, 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:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 99 , 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

Gary M. Holahan

Gary M. Holahan, Assistant Director
for Regions III and V
Division of Reactor Projects - III, IV, V
& Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 29, 1988

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 99 FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change. Corresponding overleaf pages are provided to maintain document conpleteness.

REMOVE	INSERT
3/4 1-22	3/4 1-22
3/4 5-5	3/4 5-5
3/4 5-8	3/4 5-8
3/4 7-21	3/4 7-21

POSITION INDICATOR CHANNELS-OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 All shutdown and control rod position indicator channels and the demand position indication system shall be OPERABLE and capable of determining the control rod positions within ± 12 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one rod position indicator channel per group inoperable either:

   1. Determine the position of the non-indicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and immediately after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

   2. Reduce THERMAL POWER TO < 50% of RATED THERMAL POWER within 8 hours.

b. With a maximum of one demand position indicator per bank inoperable either:

   1. Verify that all rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 12 steps of each other at least once per 8 hours, or

   2. Reduce THERMAL POWER to < 50% of RATED THERMAL POWER within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each rod position indicator channel shall be determined to be OPERABLE by verifying the demand position indication system and the rod position indicator channels agree within 12 steps at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the demand position indication system and the rod position indicator channels at least once per 4 hours.

POSITION INDICATOR CHANNELS-SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 At least one rod position indicator channel (excluding demand position indication) shall be OPERABLE for each shutdown or control rod not fully inserted.

APPLICABILITY: MODES 3*#, 4*# and 5*#

ACTION:

With less than the above required position indicator channel(s) OPERABLE, immediately open the reactor trip system breakers.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each of the above required rod position indicator channel(s) shall be determined to be OPERABLE by performance of a CHANNEL FUNCTIONAL TEST at least once per 18 months.+

* With the reactor trip system breakers in the closed position.

See Special Test Exception 3.10.5.

+ The provisions of Specification 4.0.7 are applicable.

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months by:

1. Verifying automatic isolation and interlock action of the RHR system from the Reactor Coolant System when the Reactor Coolant System pressure is above 600 psig.*

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

e. At least once per 18 months, during shutdown, by:*

1. Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection test signal.

2. Verifying that each of the following pumps start automatically upon receipt of a safety injection test signal:

a) Centrifugal charging pump

b) Safety injection pump

c) Residual heat removal pump

f. By verifying that each of the following pumps develops the indicated discharge pressure on recirculation flow when tested pursuant to Specification 4.0.5:

1. Centrifugal charging pump $\geq$ 2405 psig

2. Safety Injection pump $\geq$ 1445 psig

3. Residual heat removal pump $\geq$ 195 psig

g. By verifying the correct position of each mechanical stop for the following Emergency Core Cooling System throttle valves:

1. Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS subsystems are required to be OPERABLE.

* The provisions of Specification 4.0.7 are applicable.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.*,**

4.5.3.2 All charging pumps and safety injection pumps, except the above required OPERABLE charging pump, shall be demonstrated inoperable, by verifying that the motor circuit breakers have been removed from their electrical power supply circuits, at least once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to 152°F as determined at least once per hour when any RCS cold leg temperature is between 152°F and 200°F.

*The provisions of Specification 4.0.6 are applicable.

**The provisions of Specification 4.0.7 are applicable.

b. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are secure, and (3) in those locations where snubber movement can be manually induced without disconnecting the snubber, that the snubber has freedom of movement and is not frozen up. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specification 4.7.7.1.d as applicable. However, when the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined inoperable and cannot be determined OPERABLE via functional testing for the purpose of establishing the next visual inspection interval. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

c. Functional Tests+

At least once per 18 months during shutdown, a representative sample (10%) of the total of each type of snubber in use in the plant shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of Specification 4.7.7.1.d an additional 10% of that type of snubber shall be functionally tested.

The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers. At least 25% of the snubbers in the representative sample shall include snubbers from the following three categories:

1. The first snubber away from each reactor vessel nozzle
2. Snubbers within 5 feet of heavy equipment (valve, pump, turbine, motor, etc.)
3. Snubbers within 10 feet of the discharge from a safety relief valve

Snubbers identified in Table 3.7-9 as "Especially Difficult to Remove" or in "High Radiation Zones During Shutdown" shall also be included in the representative sample.*

*Permanent or other exemptions from functional testing for individual snubbers in these categories may be granted by the Commission only if a justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber operability for all design conditions at either the completion of their fabrication or at a subsequent date.
+ The provisions of Specification 4.0.7 are applicable.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 99 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 2

DOCKET NO. 50-316

1.0 INTRODUCTION

By letter dated January 11, 1988, the Indiana Michigan Power Company submitted a request for revision of the Technical Specifications (TSs), Appendix A to Facility Operating License DPR-74 for D. C. Cook Nuclear Plant, Unit 2. The proposed revision would extend the surveillance requirements for several items starting from March 2, 1988, to the next refueling outage currently scheduled to begin June 10, 1988. This one-time extension was requested due to operation at 80% of rated thermal power and various unanticipated outages of up to 49-day duration which resulted in a lower rate of fuel burnup. The affected TSs include:

TSs Affected	Description of Change
(1) 4.5.2.d.1 4.5.3.1	Delay RHR auto-closure interlock testing
(2) 4.7.7.1	Delay steam generator snubber functional testing
(3) 4.1.3.3	Delay analog rod position indication functional testing
(4) Table 4.3-1, Items 7 & 8 4.3.2.1.2 Table 4.3-2, Item 4.d Table 4.3-10, Items 2, 3, & 11	Delay RTD calibrations
(5) Table 4.3-2, Items 1.a, 2.a 3.a.1, 3.b.1, 3.c.1, 4.a	Delay testing of ESF manual actuation switches

(6) Table 4.3-1, Items 7, 9 & 10; Table 4.3-2, Item 1.d; 4.3.2.1.2; Table 4.3-6, Item 2; 4.4.11.1.b — Delay pressurizer pressure calibrations, interlock function testing, and PORV calibrations

In addition to the surveillance interval extensions, the amendment also proposes two minor editorial changes to correct errors in the present TS pages. The first of these changes adds the word "by" between the words "OPERABLE" and "the" in TS 4.3.2.1.1. The second change deletes a redundant "the" from TS 4.5.2.g. These changes are purely editorial in nature.

The Commission's staff has determined that failure to act in a timely manner to grant extensions for the residual heat removal (RHR) auto-closure interlock testing, steam generator snubber functional testing and rod position indication functional testing would result in shutdown of Unit 2. Therefore, this Safety Evaluation deals only with these three extensions. The remaining three extension requests will be evaluated by the staff at a later date under a separate cover.

2.0 DISCUSSION

The proposed amendment is the second of two submittals that request surveillance interval extensions for Unit 2, Cycle 6. The changes requested in this proposed amendment supplement the extension requests submitted in the licensee's amendment request dated October 28, 1987. Those changes were granted by the Commission's staff by Amendment No. 97 to Facility Operating License No. DPR-74. In addition, this request for surveillance extension is very similar to a recent extension which was granted for D. C. Cook, Unit 1. The reasons for the extension and the equipment included in this request are similar. In discussions between the licensee and the staff, the licensee has stated that the results of the Unit 1 extensions have been reviewed and no problems were discovered with operability or instrument drift for any of the items that are being requested for this Unit 2 extension. The specific TS changes are addressed below.

(1) RHR Auto-closure interlock

The proposed amendment requests a 4-month extension for the RHR auto-closure interlock test required by TS 4.5.2.d.1. An extension is also requested for TS 4.5.3.1 since it references TS 4.5.2. The RHR auto-closure interlock automatically isolates the RHR system from the Reactor Coolant System (RCS) if RCS pressure is above 600 psig. In order to demonstrate operability of the auto-closure interlock, it is necessary to open the RHR isolation valves in the cooldown line from the hot leg in order to verify that the valves would automatically close with the RCS pressure above 600 psig. This cannot be accomplished with the unit operating (i.e., with the RCS fully pressurized) because it would result in exposing the RHR system to pressures higher than the RHR safety valve setpoints.

Previous surveillance testing by the licensee has demonstrated that the auto-closure interlock is very reliable. The previous test results give the staff no reason to believe the auto-closure interlock would be inoperable during the extension period. The calibration for the RCS wide-range pressure transmitters, which provide input into the interlock, can be done at power and will be performed by the March 2, 1988, due date. Thus, the only portion of the interlock for which surveillances will not be current is the portion from the bistable of the RHR suction valves through valve operation. To meet the single failure criterion, all active components of the RHR System, including isolation valves, are duplicated. Therefore, any undetected failure which might result from the lengthening of the surveillance interval will likely be offset by this built-in redundancy. Also, the general fail-safe design of the systems offers an additional level of protection. Therefore, the Commission's staff finds the licensee's request for one-time surveillance extension acceptable.

(2) Steam Generator Snubbers

The proposed amendment would delay functional testing of steam generator snubbers required by TS 4.7.7.1.c. The extension is needed from March 9, 1988, until the refueling outage. The steam generator snubbers for which an extension is being requested are those numbered 91 and 92 in TS Table 3.7-9. The extension is requested because these snubbers are inaccessible during power operation and because TS 4.7.7.1.c specifically requires the testing to be performed during shutdown. Both snubbers required to be tested were selected randomly, i.e., neither of them are being tested as a result of a previous failure. Thirteen of the 32 snubbers in Units 1 and 2 have been functionally tested, and of the 13 tested only one failed, that being a failure to lock-up in compression. The problem was not generic, and the snubber passed the subsequent retest in 1985.

Visual inspection of the steam generator snubbers per TS 4.7.7.1.a is not required until after the scheduled outage start date, and for this reason, no extension for TS 4.7.7.1.a is requested. Visual inspections have been performed on steam generator snubbers at the Cook Nuclear Plant since 1975. These inspections are performed at least once per refueling cycle. No problem or potential problem has been revealed by these inspections. All snubbers have been found to be acceptable and no generic problems have developed.

A similar request for an extension for Unit 1 snubber surveillances was approved by the Commission on December 20, 1986, via Amendment 100 to the Unit 1 TSs. The Safety Evaluation for that amendment required that the snubber functional testing surveillance requirements be revised to increase the snubber testing sample size at least in proportion to the increase in the length of the refueling cycle beyond 18 months. The licensee intends to impose this requirement on themselves for the Unit 2 steam generator snubbers as well. This will require the licensee to perform functional testing of at least one more steam generator snubber during the upcoming refueling outage.

On the basis of the history of D. C. Cook Unit 2 snubber testing and inspection results, there is high confidence in the operability of the D. C. Cook Unit 2 snubbers, and operation for approximately four additional months past the due date for snubber functional testing will not result in a significant decrease in plant safety. Therefore, plant shutdown to perform snubber functional testing at the due dates indicated above would be unwarranted and the licensee's requested extension is acceptable.

(3) Rod Position Indication System

The proposed change would delay functional testing of the rod position indicator (RPI) channels required every 18 months by TS 4.1.3.3. The extension is needed from March 21, 1988, until the refueling outage expected in June 1988. Although TS 4.1.3.3 is only applicable in Modes 3, 4, and 5, the licensee believes relief is needed from this TS to continue operation in Modes 1 and 2 since TS 3/4.1.3.2 requires the RPI channels to be operable in these modes.

The surveillance the licensee performs to satisfy TS 4.1.3.3 is far more stringent than the channel functional testing that the TS requires. The test is actually a calibration of the RPI channels over the rod insertion range. Since rods must be inserted to perform the calibration, it cannot be performed at power because to do so would violate the rod insertion limits of TS 3.1.3.6. The operability of the RPI channels is functionally verified once per 12 hours per TS 4.1.3.2 by comparison to the demand position indication system. These comparisons would be expected to indicate significant degradation in the RPI channels. Indication that the core is performing as designed is provided by the incore flux maps, which are taken at least once every 31 effective full power days to satisfy the requirements of TSs 4.2.2.2 ($F_Q(Z)$) and 4.2.3 ($F^N_{\Delta H}$). Core performance is also indicated by the excore detectors, which are used to measure the quadrant power tilt ratio per TS 4.2.4. These surveillances would also be expected to indicate discrepancies between indicated and actual rod position.

Therefore, the staff finds the licensee's request for a one-time surveillance extension acceptable.

3.0 EMERGENCY CIRCUMSTANCES

The Commission has determined that emergency circumstances exist in that swift action is necessary to avoid shutdown of D. C. Cook Unit 2. The need for certain surveillance extensions was identified in September 1987. The dates that the surveillance intervals would expire fell into two groups. Several were due during the brief period of time from December 31, 1987, through January 4, 1988. The second group was not due until the period beginning March 2, 1988, and continuing into the Unit 2 refueling outage. Since there was a gap of approximately two months between the groups, in discussions between the staff and the licensee, the decision was made that two requests for surveillance extensions would be appropriate. Thus, attention could be focused on those requests which involved the most immediate need. The licensee's January 11, 1988, proposed amendment is the second of the two submittals that requested surveillance interval extensions for Unit 2, Cycle 6. The changes requested in the January 11, 1988 letter supplement the extensions requested by the letter dated October 28, 1987. Those changes were granted by the Commission by Amendment No. 97 to Facility Operating License No. DPR-74 dated December 28, 1987.

Inadvertently the licensee's January 11, 1988 request was not promptly noticed by the NRC. Notice requesting comments on the Commission's proposed no significant hazards consideration determination was published in the Federal Register on February 17, 1988 (53 FR 4796). No comments have been received.

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards considerations if operation of the facility, in accordance with the amendment, would not:

(1) Involve a significant increase in the probability or consequences of any 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 requested amendment has been evaluated against the standards in 10 CFR 50.92 as follows:

(1) RHR Auto-closure Interlock Test

Criterion 1

The surveillance test history of the auto-closure interlock has shown that the system is highly reliable, and there is no reason to believe the equipment would be inoperable during an extension period. The wide-range pressure transmitters, which provide input into the auto-closure interlock, will have a current calibration. Additionally, when the RHR system is not in service, power is removed from the suction valve operators, thus preventing inadvertent valve opening and eliminating the need for the auto-closure interlock. For these reasons, the extension is not expected to result in a significant increase in the probability or consequences of a previously evaluated accident, nor will it result in a significant reduction in a margin of safety.

Criterion 2

This extension will not result in a change in plant configuration or operation. Therefore, the extension should not create the possibility of a new or different kind of accident from any previously evaluated or analyzed.

Criterion 3

See Criterion 1, above.

(2) Steam Generator Snubbers

Criterion 1

Thirteen steam generator snubbers have been functionally tested at the Cook Nuclear Plant since 1983 with only one failure, the cause of which was not generic. Visual inspections have been performed on snubbers since 1975, revealing no problems or potential problems. Based on this surveillance history, the steam generator snubbers are not expected to be inoperable during the extension period. Thus, it is believed that this change will not result in a significant increase in the probability or consequences of a previously evaluated accident, nor will it significantly reduce a margin of safety.

Criterion 2

Delaying the snubber functional test will not result in a change in plant design or operation. Therefore, the change will not create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated.

Criterion 3

See Criterion 1, above.

(3) Rod Position Indication System Functional Testing

Criterion 1

TS-required comparison of the RPI channels to the demand position indication system would be expected to indicate significant degradation in the RPI channels. In addition, other surveillance, such as the determination of the quadrant power tilt ratio and incore flux mapping, provide a comparison of core performance to design and would be expected to indicate significant deviations of the control rods from their indicated position. Also, the RPI channel surveillance history is good and provides no reason to believe the changes would be inoperable during the extension period. For all these reasons, the change will not involve a significant increase in the probability or consequences of a previously analyzed accident and it will not involve a significant reduction in a margin of safety.

Criterion 2

The proposed change will not result in a change in plant configuration or operation. Thus, the change should not create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated.

Criterion 3

See Criterion 1, above.

Therefore, based on these considerations and the three standards given above, the Commission has made a final determination that the requested changes involve no significant hazards consideration.

5.0 STATE CONSULATION

In accordance with the Commission's regulations, efforts were made to contact the Michigan representative. The state representative was contacted and had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes in the 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 made a final no significant hazards consideration finding with respect to this amendment. Accordingly, this 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 this amendment.

7.0 CONCLUSION

The staff 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, 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: February 29, 1988

Principal Contributor: John F. Stang