

August 20, 1990

Docket Nos. 50-315
and 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: AMENDMENT NOS. 146 AND 133 TO FACILITY OPERATING LICENSE NOS. DPR-58
AND DPR-74: (TAC NO. 76650)

The Commission has issued the enclosed Amendment No. 146 to Facility Operating License No. DPR-58 and Amendment No. 133 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Units Nos. 1 and 2. The amendments consist of changes to the Technical Specifications in response to your application dated April 16, 1990.

These amendments change the Unit 1 Technical Specifications (TS) by removing cycle-specific core operating limits from the TS and placing them into a cycle-specific Core Operating Limits Report (COLR). This change is in accordance with NRC Generic Letter 86-16. Additionally, surveillance requirement 4.1.1.4.1, which addresses measurement of the moderator temperature coefficient (MTC), is removed from the TS of both units as this requirement is specifically addressed in the COLR submitted for each unit.

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

Sincerely,



Timothy Colburn, Project Manager
Project Directorate III-1
Division of Reactor Projects - III,
IV, V & Special Projects
Office of Nuclear Reactor Regulation

Enclosures:

1. Amendment No. 146 to DPR-58
2. Amendment No. 133 to DPR-74
3. Safety Evaluation

cc w/enclosures:
See next page

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** subject to changes J.M.*



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P FDC



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

August 20, 1990

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and 50-316

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Indiana Michigan Power Company
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1 Riverside Plaza
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A copy of our related Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Handwritten signature of Timothy M. Colburn in cursive.

Timothy Colburn, Project Manager
Project Directorate III-1
Division of Reactor Projects - III,
IV, V & Special Projects
Office of Nuclear Reactor Regulation

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cc w/enclosures:
See next page

Mr. Milton Alexich
Indiana Michigan Power Company

Donald C. Cook Nuclear Plant

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

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 146
License No. DPR-58

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Indiana Michigan Power Company (the licensee) dated April 16, 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.

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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-58 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 146, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Dominic C. D. Danni/RB

Robert Pierson, Director
Project Directorate III-1
Division of Reactor Projects - III,
IV, V & Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 20, 1990



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. 133
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 April 16, 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. 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. 133, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Dominic C. D. Lanni / R.P.

Robert Pierson, Director
Project Directorate III-1
Division of Reactor Projects - III,
IV, V & Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Charges to the Technical
Specifications

Date of Issuance: August 20, 1990

ATTACHMENT TO LICENSE AMENDMENT NO.146

FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

REMOVE

II
1-7
3/4 1-5
3/4 1-5a
-
3/4 1-19
3/4 1-21 through 24
3/4 2-1
3/4 2-3
3/4 2-4
3/4 2-5
3/4 2-8
3/4 2-9
3/4 2-15
6-19
-
B 2-2
B 2-2a
B 3/4 2-1 through 4

INSERT

II
1-7
3/4 1-5
3/4 1-5a
3/4 1-5b
3/4 1-19
3/4 1-21 through 24
3/4 2-1
3/4 2-3
3/4 2-4
3/4 2-5
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DEFINITIONS

MEMBER(S) OF THE PUBLIC

1.35 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or its vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant.

SITE BOUNDARY

1.36 The SITE BOUNDARY shall be that line beyond which the land is not owned, leased or otherwise controlled by the licensee.

UNRESTRICTED AREA

1.37 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY to which access is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials or any area within the site boundary used for residential quarters or industrial, commercial, institutional and/or recreational purposes.

ALLOWABLE POWER LEVEL (APL)

1.38 APL means "allowable power level" which is that power level, less than or equal to 100% RATED THERMAL POWER, at which the plant may be operated to ensure that power distribution limits are satisfied.

CORE OPERATING LIMITS REPORT (COLR)

1.39 The COLR is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.11. Unit operation within these operating limits is addressed in individual specifications.

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be within the limits specified in the COLR. The maximum upper limit shall be less than or equal to the limit shown in Figure 3.1-2.

APPLICABILITY: BOL Limit - MODES 1 and 2* only#
EOL Limit - MODES 1, 2 and 3 only#

ACTION:

- a. With the MTC more positive than the BOL limit specified in the COLR:
1. Establish and maintain control rod withdrawal limits sufficient to restore the MTC to within its limit within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.5.
 2. Maintain the control rods within the withdrawal limits established above until subsequent measurement verifies that the MTC has been restored to within its limit for the all rods withdrawn condition.
 3. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 10 days describing the value of the measured MTC, the interim control rod withdrawal limits and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.
- b. With the MTC more negative than the EOL limit specified in the COLR, be in HOT SHUTDOWN within 12 hours.

* With K_{eff} greater than or equal to 1.0
See Special Test Exception 3.10.4

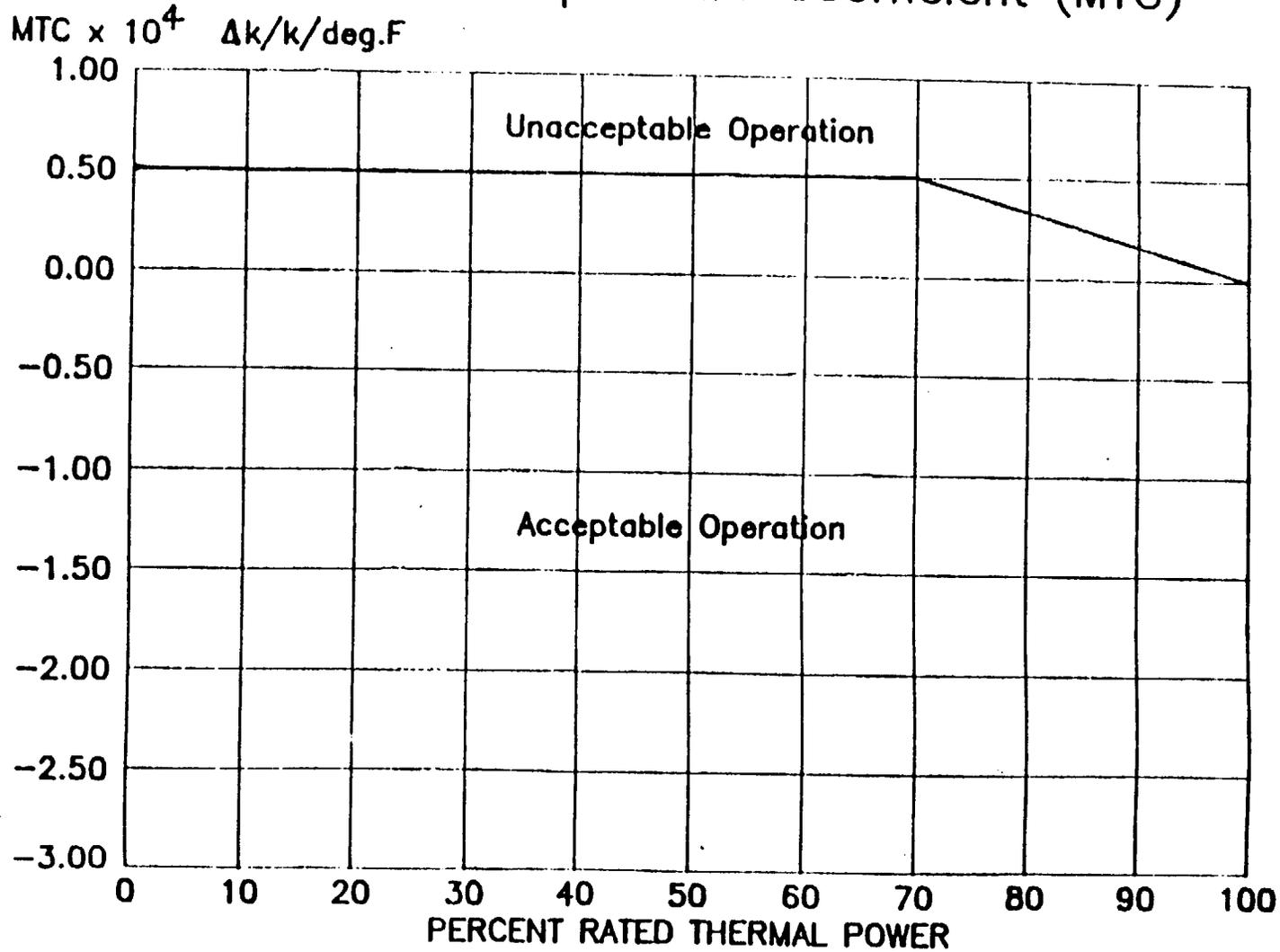
REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS

- 4.1.1.4 The MTC shall be determined to be within its limits during each fuel cycle as follows:
- a) The MTC shall be measured and compared to the BOL limit specified in the COLR prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.
 - b) The MTC shall be measured at any THERMAL POWER within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. The measured value shall be compared to the 300 ppm surveillance limit specified in the COLR. In the event this comparison indicates that the MTC will be more negative than the EOL limit, the MTC shall be remeasured at least once per 14 EFPD during the remainder of the fuel cycle and the MTC value compared to the EOL limit.

FIGURE 3.1-2

Moderator Temperature Coefficient (MTC)



REACTIVITY CONTROL SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

- c) A power distribution map is obtained from the movable incore detectors and $F_0(Z)$ and $F_{\Delta H}^N$ are verified to be within their limits within 72 hours, and
- d) Either the THERMAL POWER level is reduced to less than or equal to 75% of RATED THERMAL POWER within one hour and within the next 4 hours the high neutron flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER, or
- e) The remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod within one hour while maintaining the rod sequence and insertion limits as specified in the COLR; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.5 during subsequent operation.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full length rod not fully inserted shall be determined to be OPERABLE by movement of at least 8 steps in any one direction at least once per 31 days.

REACTIVITY CONTROL SYSTEMS

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.3 The individual full length (shutdown and control) rod drop time from the fully withdrawn position (specified in the COLR) shall be less than or equal to 2.4 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

- a. T_{avg} greater than or equal to 541^oF, and
- b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 AND 2

ACTION:

With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

SURVEILLANCE REQUIREMENTS

4.1.3.3 The rod drop time of full length rods shall be demonstrated through measurement prior to entering MODE 2:

- a. For all rods following each removal of the reactor vessel head,
- b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and
- c. At least once per 18 months.

REACTIVITY CONTROL SYSTEMS

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.4 All shutdown rods shall be limited in physical insertion as specified in the COLR.

APPLICABILITY: MODES 1* and 2*#

ACTION:

With a maximum of one shutdown rod inserted beyond the insertion limit specified in the COLR, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

- a. Restore the rod to within the insertion limit specified in the COLR,
or
- b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.4 Each shutdown rod shall be determined to be within the insertion limit specified in the COLR:

- a. Within 15 minutes prior to withdrawal of any rods in control banks A, B, C or D during an approach to reactor criticality,
and
- b. At least once per 12 hours thereafter.

* See Special Test Exceptions 3.10.2 and 3.10.4.

With K_{eff} greater than or equal to 1.0

REACTIVITY CONTROL SYSTEMS

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.5 The control banks shall be limited in physical insertion as specified in the COLR.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the control banks inserted beyond the insertion limits, except for surveillance testing pursuant to Specification 4.1.3.1.2; either:

- a. Restore the control banks to within the limits within two hours,
or
- b. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the group position using the insertion limits specified in the COLR, or
- c. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.5 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

* See Special Test Exceptions 3.10.2 and 3.10.4

With K_{eff} greater than or equal to 1.0.

Figure 3.1-1 intentionally deleted.

3/4.2 POWER DISTRIBUTION LIMITS

AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within the target band about a target flux difference. The target band is specified in the COLR.

APPLICABILITY: MODE 1 above 50% RATED THERMAL POWER*

ACTION:

- a. With the indicated AXIAL FLUX DIFFERENCE outside of the target band about the target flux difference and with THERMAL POWER:
 1. Above 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER, within 15 minutes:
 - a) Either restore the indicated AFD to within the target band limits, or
 - b) Reduce THERMAL POWER to less than 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER.
 2. Between 50% and 90% or 0.9 x APL (whichever is less) of RATED THERMAL POWER:
 - a) POWER OPERATION may continue provided:
 - 1) The indicated AFD has not been outside of the target band for more than 1 hour penalty deviation cumulative during the previous 24 hours, and
 - 2) The indicated AFD is within the limits specified in the COLR. Otherwise, reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.
 - b) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1.1 provided the indicated AFD is maintained within the limits specified in the COLR. A total of 16 hours operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

* See Special Test Exception 3.10.2

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS (Continued)

4.2.1.2 The indicated AFD shall be considered outside of its target band when at least 2 of 4 or 2 of 3 OPERABLE excore channels are indicating the AFD to be outside the target band. Penalty deviation outside of the target band shall be accumulated on a time basis of:

- a. A penalty deviation of one minute for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and
- b. A penalty deviation of one-half minute for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER.

4.2.1.3 The target axial flux difference of each OPERABLE excore channel shall be determined in conjunction with the measurement of APL as defined in Specification 4.2.6.2. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 The axial flux difference target band about the target axial flux difference shall be determined in conjunction with the measurement of APL as defined in Specification 4.2.6.2. The allowable values of the target band are specified in the COLR. The provisions of Specification 4.0.4 are not applicable.

Figure 3.2-1 intentionally deleted.

POWER DISTRIBUTION LIMITS

HEAT FLUX HOT CHANNEL FACTOR-F_Q(Z)

LIMITING CONDITION FOR OPERATION

3.2.2 F_Q(Z) shall be limited by the following relationships:

$$F_Q(Z) \leq CFQ/P [K(Z)] \quad P > 0.5$$

$$F_Q(Z) \leq CFQ/0.5 [K(Z)] \quad P \leq 0.5$$

- o CFQ is the F_Q limit at RATED THERMAL POWER specified in the COLR
- o $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$
- o F_Q(Z) is the measured hot channel factor including a 3% manufacturing tolerance uncertainty and a 5% measurement uncertainty.
- o K(Z) is the normalized F_Q(Z) as a function of core height specified in the COLR.

APPLICABILITY: MODE 1

ACTION:

With F_Q(Z) exceeding its limit:

- a. Reduce THERMAL POWER at least 1% for each 1% F_Q(Z) exceeds the limit within 15 minutes and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower ΔT Trip Setpoints have been reduced at least 1% for each 1% F_Q(Z) exceeds the limit.
- b. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a, above; THERMAL POWER may then be increased provided F_Q(Z) is demonstrated through incore mapping to be within its limit.

Figure 3.2-3 intentionally deleted.

POWER DISTRIBUTION LIMITS

NUCLEAR ENTHALPY HOT CHANNEL FACTOR - $F_{\Delta H}^N$

LIMITING CONDITION FOR OPERATION

3.2.3 $F_{\Delta H}^N$ shall be limited by the following relationship:

$$F_{\Delta H}^N \leq \text{CFDH} [1 + \text{PFDH} (1-P)]$$

where: P is the fraction of RATED THERMAL POWER

CFDH is the $F_{\Delta H}^N$ limit at RATED THERMAL POWER specified in the COLR

PFDH is the power factor multiplier for $F_{\Delta H}^N$ specified in the COLR

APPLICABILITY: MODE 1

ACTION:

With $F_{\Delta H}^N$ exceeding its limit:

- a. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours,
- b. Demonstrate through in-core mapping that $F_{\Delta H}^N$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours, and
- c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION may proceed, provided that $F_{\Delta H}^N$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

POWER DISTRIBUTION LIMITS

ALLOWABLE POWER LEVEL - APL

LIMITING CONDITION FOR OPERATION

3.2.6 THERMAL POWER shall be less than or equal to ALLOWABLE POWER LEVEL (APL), given by the following relationship:

$$\text{APL} = \min \text{ over } Z \text{ of } \frac{\text{CFQ} \times \text{K}(Z)}{\text{F}_Q(Z) \times \text{V}(Z) \times \text{F}_p} \times 100\%, \text{ or } 100\%, \text{ whichever is less}$$

- o CFQ is the F_Q limit at RATED THERMAL POWER specified in the COLR for Westinghouse or Exxon fuel.
- o $\text{F}_Q(Z)$ is the measured hot channel factor including a 3% manufacturing tolerance uncertainty and a 5% measurement uncertainty.
- o $\text{V}(Z)$ is the function specified in the COLR.
- o $\text{F}_p = 1.00$ except when successive steady-state power distribution maps indicate an increase in max over Z of $\frac{\text{F}_Q(Z)}{\text{K}(Z)}$ with exposure.

Then either of the penalties, F_p , shall be taken:

$$\text{F}_p = 1.02 \text{ or,}$$

$\text{F}_p = 1.00$ provided that Surveillance Requirement 4.2.6.2 is satisfied once per 7 Effective Full Power Days until two successive maps indicate that the max over Z of $\frac{\text{F}_Q(Z)}{\text{K}(Z)}$ is not increasing.

- o The above limit is not applicable in the following core regions.
 - 1) Lower core region 0% to 10% inclusive.
 - 2) Upper core region 90% to 100% inclusive.

APPLICABILITY: MODE 1

ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT

6.9.1.11.1 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

- a. Moderator Temperature Coefficient Limits for Specification 3/4.1.1.4.
- b. Rod Drop Time Limits for Specification 3/4.1.3.3,
- c. Shutdown Rod Insertion Limits for Specification 3/4.1.3.4,
- d. Control Rod Insertion Limits for Specification 3/4.1.3.5,
- e. Axial Flux Difference for Specification 3/4.2.1,
- f. Heat Flux Hot Channel Factor for Specification 3/4.2.2,
- g. Nuclear Enthalpy Rise Hot Channel Factor for Specification 3/4.2.3, and
- h. Allowable Power Level for Specification 3/4.2.6.

6.9.1.11.2 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

- a. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 (Westinghouse Proprietary),
- b. WCAP-8385, "Power Distribution Control and Load Following Procedures - Topical Report," September 1974 (Westinghouse Proprietary),
- c. WCAP-10216-P-A, Part B, "Relaxation of Constant Axial Offset Control/ F_0 Surveillance Technical Specification," June 1983 (Westinghouse Proprietary),

ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT (Continued)

- d. WCAP-10266-P-A Rev. 2, "The 1981 Version of Westinghouse Evaluation Mode Using BASH Code," March 1987 (Westinghouse Proprietary).

6.9.1.11.3 The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.11.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC document control desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

- a. Inservice Inspection Program Review, Specification 4.4.10
- b. ECCS Actuation, Specifications 3.5.2 and 3.5.3
- c. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3
- d. Inoperable Meteorological Monitoring Instrumentation, Specification 3.3.3.4
- e. Seismic Event Analysis, Specification 4.3.3.3.2
- f. Sealed Source Leakage in Excess of Limits, Specification 4.7.7.1.3
- g. Moderator Temperature Coefficient, Specification 3.1.1.4

SAFETY LIMITS

BASES

The curves are based on a nuclear enthalpy rise hot channel factor, $F_{\Delta H}^N$, defined in the COLR and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F_{\Delta H}^N$ at reduced power based on the expression:

$$F_{\Delta H}^N = \text{CFDH} [1 + \text{PFDH} (1-P)]$$

where: CFDH is the $F_{\Delta H}^N$ limit at RTP specified in the COLR

PFDH is the power factor multiplier for $F_{\Delta H}^N$ provided in the COLR

P is the fraction of RATED THERMAL POWER

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion, assuming the axial power imbalance is within the limits of the f_1 (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant, which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B 31.1 1967 Edition, which permits a maximum transient pressure of 120% (2985 psig) of component design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrottested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

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3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core greater than or equal to the safety limit DNBR during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F_{\Delta H}^N$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

The limits on $F_Q(Z)$ and $F_{\Delta H}^N$ are specified in the COLR.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE ensure that the $F_Q(Z)$ upper bound envelope is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The $F_Q(Z)$ upper bound envelope is specified in the COLR.

Target flux difference is determined at equilibrium xenon conditions. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

POWER DISTRIBUTION LIMITS

BASES

Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the target band about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits as specified in the COLR while at THERMAL POWER levels above 50% of RATED THERMAL POWER. For THERMAL POWER levels between 15% and 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the target band and the THERMAL POWER is greater than 90% or 0.9 x APL of RATED THERMAL POWER (whichever is less). During operation at THERMAL POWER levels between 50% and 90% or 0.9 x APL of RATED THERMAL POWER (whichever is less) and between 15% and 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour and 2 hours, respectively.

The upper bound limit (90% or 0.9 x APL of RATED THERMAL POWER (whichever is less)) on AXIAL FLUX DIFFERENCE assures that the $F_0(Z)$ envelope is not exceeded during either normal operation or in the event of xenon redistribution following power changes. The lower bound limit (50% of RATED THERMAL POWER) is based on the fact that at THERMAL POWER levels below 50% of RATED THERMAL POWER, the average linear heat generation rate is half of its nominal operating value and below that value, perturbations in localized flux distributions cannot affect the results of ECCS or DNBR analyses in a manner which would adversely affect the health and safety of the public.

The bases and methodology for establishing these limits is presented in topical report WCAP-8385, "Power Distribution Control and Load Following Procedures."

Figure B 3/4 2-1 intentionally deleted.

POWER DISTRIBUTION LIMITS

BASES

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR, AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, and nuclear enthalpy rise hot channel factors ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable, but will normally only be determined periodically, as specified in Specifications 4.2.2.1, 4.2.2.2, 4.2.3, 4.2.6.1 and 4.2.6.2. This periodic surveillance is sufficient to ensure that the hot channel factor limits are maintained provided:

- a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps from the group demand position.
- b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.5.
- c. The control rod insertion limits of Specifications 3.1.3.4 and 3.1.3.5 are maintained.
- d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

The relaxation in F_{H}^{N} as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. F_{H}^{N} will be maintained within its limits as specified in the COLR, provided conditions (a) through (d) above are maintained.

When an F_{O} measurement is taken, both experimental error and manufacturing tolerance must be allowed for. 5% is the appropriate allowance for a full core map taken with the incore detector flux mapping system, and 3% is the appropriate allowance for manufacturing tolerance.

When F_{H}^{N} is measured, experimental error must be allowed for, and 4% is the appropriate allowance for a full core map taken with the incore detection system. This 4% measurement uncertainty has been included in the design DNBR limit value. The specified limit for F_{H}^{N} also contains an additional 4% allowance for uncertainties. The total allowance is based on the following considerations:

ATTACHMENT TO LICENSE AMENDMENT NO. 133

FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Revise Appendix A Technical Specifications by removing the page identified below and inserting the attached page. The revised page are identified by amendment number and contain marginal lines indicating the area of change.

REMOVE

3/4 1-6

INSERT

3/4 1-6

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS

4.1.1.4 The MTC shall be determined to be within its limits during each fuel cycle as follows:

- a) The MTC shall be measured and compared to the BOL limit specified in the COLR prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.
- b) The MTC shall be measured at any THERMAL POWER within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. The measured value shall be compared to the 300 ppm surveillance limit specified in the COLR. In the event this comparison indicates that the MTC will be more negative than the EOL limit, the MTC shall be remeasured at least once per 14 EFPD during the remainder of the fuel cycle and the MTC value compared to the EOL limit.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 146 TO FACILITY OPERATING LICENSE NO. DPR-58
AND AMENDMENT NO. 133 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By letter dated April 16, 1990, (Ref. 1), the Indiana Michigan Power Company (the licensee) proposed changes to the Technical Specifications (TS) for the Donald C. Cook Nuclear Plant, Unit No. 1. The proposed changes would modify specifications having cycle-specific parameter limits by replacing the values of those limits with a reference to the Core Operating Limits Report (COLR) for the values of those limits. The proposed changes also include the addition of the COLR to the Definitions section and to the reporting requirements of the Administrative Controls sections of the TS. Guidance on the proposed changes was developed by NRC on the basis of the review of a lead-plant proposal submitted on the Oconee plant docket by Duke Power Company. This guidance was provided to all power reactor licensees and applicants by Generic Letter 88-16, dated October 4, 1988, (Ref. 2).

In addition, the licensee proposed other Technical Specification changes not related to Generic Letter 88-16. These changes have also been reviewed.

2.0 EVALUATION

2.1 Technical Specification Changes to Implement Generic Letter 88-16

The licensee's proposed changes to the TS are in accordance with the guidance provided by Generic Letter 88-16 and are addressed below.

- (1) The Definitions section of the TS was modified to include a definition of the Core Operating Limits Report that requires cycle/reload-specific parameter limits to be established on a unit-specific basis in accordance with NRC approved methodologies that maintain the limits of the safety analysis. The definition notes that plant operation within these limits is addressed by individual specifications.

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(2) The following specifications were revised to replace the values of cycle-specific parameter limits with a reference to the COLR that provides these limits.

(a) Specification 3/4.1.1.4

The moderator temperature coefficient (MTC) limits for this specification and surveillance requirement are specified in the COLR.

(b) Specification 3/4.1.3.3

The rod drop time fully withdrawn position for this specification is specified in the COLR.

(c) Specification 3/4.1.3.4

The shutdown bank insertion limit for this specification and surveillance requirement is specified in the COLR.

(d) Specification 3/4.1.3.5

The control bank insertion limits for this specification are specified in the COLR.

(e) Specification 3/4.2.1

The axial flux difference limits for this specification and surveillance requirement are specified in the COLR.

(f) Specification 3/4.2.2

The heat flux hot channel factor (F_0) limit at rated thermal power and the normalized F_0 limit as a function of core height ($K(Z)$) for this specification are specified in the COLR.

(g) Specification 3/4.2.3

The nuclear enthalpy rise hot channel factor (F-delta-H) limit at rated thermal power and the power factor multiplier for this specification are specified in the COLR.

(h) Specification 3/4.2.6

The allowable power lever (APL) limit and the transient xenon effect on F_0 as a function of core height ($V(Z)$) for this specification are specified in the COLR.

(i) Specification 3.1.3.1

This Moveable Control Assemblies - Group Height Specification's action statement was revised to reference the control rod insertion limits specified in the COLR.

The affected bases of the specifications have been modified by the licensee to include appropriate reference to the COLR. Based on our review, we conclude that the changes to these bases are acceptable.

- (3) Specification 6.9.1.11 was added to the reporting requirements of the Administrative Controls section of the TS. This specification requires that the COLR be submitted, upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector. The report provides the values of cycle-specific parameter limits that are applicable for the current fuel cycle. Furthermore, these specifications require that the values of these limits be established using NRC approved methodologies and be consistent with all applicable limits of the safety analysis. The approved methodologies are the following:

- (a) WCAP-9727-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985 (Westinghouse Proprietary),
- (b) WCAP-8385, "Power Distribution Control and Load Following Procedures - Topical Report," September 1974 (Westinghouse Proprietary),
- (c) WCAP-10216-P-A, Part B, "Relaxation of Constant Axial Offset Control/ F_0 Surveillance Technical Specification," June 1983 (Westinghouse Proprietary),
- (d) WCAP-10266-P-A Rev. 2, "The 1981 Version of Westinghouse Evaluation Model Using BASH Code," March 1987 (Westinghouse Proprietary).

Finally, the specification requires that all changes in cycle-specific parameter limits be documented in the COLR before each reload cycle or remaining part of a reload cycle and submitted upon issuance to NRC, prior to operation with the new parameter limits.

On the basis of the review of the above items, the NRC staff concludes that the licensee provided an acceptable response to those items as addressed in the NRC guidance in Generic Letter 88-16 on modifying cycle-specific parameter limits in the TS. Because plant operation continues to be limited in accordance with the values of cycle-specific parameter limits that are established using NRC approved methodologies, the NRC staff concludes that this change is administrative in nature and there is no impact on plant safety as a consequence. Accordingly, the staff finds that the proposed changes are acceptable.

As part of the implementation of Generic Letter 88-16, the staff has also reviewed a sample COLR that was provided by the licensee. On the basis of this review, the staff concludes that the format and content of the sample COLR are acceptable.

2.2 Other Technical Specification Changes

In addition to the TS changes to implement a COLR, the licensee requested a number of other TS changes. Changes were requested to the action statement and surveillance requirements of TS 3/4.1.1.4 regarding the moderator temperature coefficient (MTC). The action statement is being elaborated and the surveillance requirements are being revised to incorporate a 300 ppm surveillance acceptance criterion by reference to the COLR. The use of the surveillance criterion at 300 ppm and the direct comparison of the beginning of life (BOL) surveillance to test results permits the deletion of Surveillance Requirement 4.1.1.4.1, which previously had required an extrapolation in order to permit direct comparison with test results. Also, the deletion of Surveillance Requirement 4.1.1.4.1 is acceptable because the comparison of calculations with test results would of necessity include the proper extrapolations and compensations in order to allow for a valid comparison. Therefore, this change is of minor safety significance. Based on our review, we conclude that these changes to TS 3/4.1.1.4 are acceptable for D.C Cook Unit 1. We note that the changes to implement a COLR for D. C. Cook Unit 2 were approved in amendment No. 122 issued on May 23, 1990. We also conclude that the deletion of Surveillance Requirement 4.1.1.4.1 for D. C. Cook, Unit 2 is acceptable for the reasons outlined above for D. C. Cook, Unit 1.

TS 3.1.3.1 on Moveable Control Assemblies - Group Height is being revised by removing the reference to a figure on rod bank step position as a function of power and referencing the COLR. This change is administrative and, therefore, is acceptable.

TS 6.9.2 on Special Reports is being revised so that 6.9.2g refers to the MTC special report required by TS 3.1.1.4. In addition, Item 6.9.2h which states "Deleted" is being deleted. These changes are administrative in nature and, therefore, are acceptable.

We have reviewed the request by the Indiana and Michigan Power Company to modify the Technical Specifications of the Donald C. Cook Nuclear Plant Unit No. 1 that would remove the specific values of some cycle-dependent parameters from the specifications and place the values in a Core Operating Limits Report that would be referenced by the specifications. Based on this review, we conclude that these Technical Specification modifications are acceptable and are in accordance with the provisions of Generic Letter 88-16.

We have also reviewed non-Core Operating Limits Report TS changes and conclude that these changes are acceptable. In particular, we conclude that deletion of Surveillance Requirement 4.1.1.4.1 for D. C. Cook, Unit 2 is also acceptable.

3.0 ENVIRONMENTAL CONSIDERATION

These amendments involve changes in a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. We have determined that the amendments involve 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 these amendments involve no significant hazards consideration and there has been no public comment on such finding. In addition, these amendments involve changes to reporting or recordkeeping requirements. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

4.0 CONCLUSION

We have concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, (3) and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

5.0 REFERENCES

1. Letter (AEP:NRC:1077B) from M. P. Alexich (IMPC) to NRC, dated April 16, 1990.
2. Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," dated October 4, 1988.

Date: August 20, 1990

Principal Contributor: D. Fieno, SRXB