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August 10, 2005

AEP:NRC:5331
10 CFR 50.90

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

SUBJECT: Donald C. Cook Nuclear Plant Units 1 and 2
Docket Nos. 50-315 and 50-316
Deletion of the Power Range Neutron Flux High Negative Rate Trip Function

References:

1. Westinghouse Topical Report, WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," dated January 1990.

2. Letter from A. C. Thadani, Nuclear Regulatory Commission (NRC), to R. A. Newton Westinghouse Owners Group, "Acceptance for Referencing of Licensing Topical Reports WCAP-11394(P) and WCAP-11395(NP), 'Methodology for the Analysis of the Dropped Rod Event'," dated October 23, 1989.

3. Letter from J. Donohew, NRC, to M. K. Nazar, Indiana Michigan Power Company, "D.C. Cook Nuclear Plant, Units 1 and 2 – Issuance of Amendments for the Conversion to the Improved Technical Specifications with Beyond Scope Issues (TAC NOS. MC2629, MC2630, MC2653 THROUGH MC2687, MC2690 through MC2695, MC3152 thorugh MC3157, MC3432 through MC3453)," dated June 1, 2005.

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP) Units 1 and 2, proposes to amend Facility Operating Licenses DPR-58 and DPR-74. I&M proposes to modify Technical Specifications (TS) to delete the power range neutron flux high negative rate trip. The proposed changes are consistent with the methodology presented in the Westinghouse Topical Report WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," Reference 1, as accepted by the Nuclear Regulatory Commission (NRC) in Reference 2. The NRC has previously approved similar TS amendments at Watts Bar, Braidwood/Byron, and Seabrook nuclear plants (Accession Numbers ML020780104, ML011410291, and ML032310339) on January 15, 1999, May 17, 2001, and October 1, 2003, respectively.

A001

By Reference 2, the NRC approval of WCAP-11394-P-A stated, "A further review by the staff (for each cycle) is not necessary, given the utility assertion that the analysis described by Westinghouse has been performed and the required comparisons have been made with favorable results." For the past several fuel cycle designs, a dropped Rod Cluster Control Assembly (RCCA) analysis has been performed in accordance with the methodology described in WCAP-11394-P-A. Performance of the dropped RCCA analysis for future fuel cycle designs will be formalized in the CNP Nuclear Fuel administrative procedure for core designs.

By Reference 3, NRC approved I&M's conversion of the CNP Current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) specified in NUREG-1431, "Standard Technical Specifications – Westinghouse Plants," Revision 2. I&M intends to implement ITS no later than October 31, 2005; however, ITS have not yet been implemented. I&M has therefore provided copies of both the CTS and the ITS pages that are affected by this proposed amendment. I&M will coordinate with the NRC Project Manager to ensure that the appropriate pages are issued.

Enclosure 1 provides an affirmation statement pertaining to this letter. Enclosure 2 provides I&M's evaluation of the proposed change. Attachments 1A and 1B provide CTS pages marked to show changes for Unit 1 and Unit 2, respectively. Attachments 2A and 2B provide CTS pages with the proposed changes incorporated. Attachments 3A and 3B provide ITS pages marked to show changes for Unit 1 and Unit 2, respectively. Attachments 4A and 4B provide ITS pages with the proposed changes incorporated. Attachment 5 provides the regulatory commitment made in this submittal.

I&M requests approval of the proposed amendment prior to March 1, 2006. I&M requests a 30-day implementation period following approval.

Copies of this letter and its attachments are being transmitted to the Michigan Public Service Commission and Michigan Department of Environmental Quality, in accordance with the requirements of 10 CFR 50.91.

Should you have any questions, please contact Mr. John A. Zwolinski, Safety Assurance Director at (269) 466-2428.

Sincerely,

Daniel P. Fadel
Engineering Vice President

KS/rdw

Enclosures:

1. Affirmation
2. Licensee's Evaluation

Attachments:

1A. Donald C. Cook Nuclear Plant Unit 1 Current Technical Specification Pages Marked To Show Changes
1B. Donald C. Cook Nuclear Plant Unit 2 Current Technical Specification Pages Marked To Show Changes
2A. Donald C. Cook Nuclear Plant Unit 1 Current Technical Specification Pages With the Proposed Changes Incorporated
2B. Donald C. Cook Nuclear Plant Unit 2 Current Technical Specification Pages With the Proposed Changes Incorporated
3A. Donald C. Cook Nuclear Plant Unit 1 Improved Technical Specification Pages Marked To Show Changes
3B. Donald C. Cook Nuclear Plant Unit 2 Improved Technical Specification Pages Marked To Show Changes
4A. Donald C. Cook Nuclear Plant Unit 1 Improved Technical Specification Pages With the Proposed Changes Incorporated
4B. Donald C. Cook Nuclear Plant Unit 2 Improved Technical Specification Pages With the Proposed Changes Incorporated
5. Regulatory Commitments

c: J. L. Caldwell, NRC Region III
K. D. Curry, Ft. Wayne AEP, w/o enclosures/attachments
J. T. King, MPSC
C. F. Lyon, NRC Washington, DC
MDEQ – WHMD/RPMWS
NRC Resident Inspector

Enclosure 1 to AEP:NRC:5331

AFFIRMATION

I, Daniel P. Fadel, being duly sworn, state that I am Engineering Vice President of Indiana Michigan Power Company (I&M), that I am authorized to sign and file this request with the Nuclear Regulatory Commission on behalf of I&M, and that the statements made and the matters set forth herein pertaining to I&M are true and correct to the best of my knowledge, information, and belief.

Indiana Michigan Power Company

Daniel P. Fadel
Engineering Vice President

SWORN TO AND SUBSCRIBED BEFORE ME

THIS 10 DAY OF August, 2005

Notary Public

My Commission Expires 6/10/2007

Enclosure 2 to AEP:NRC:5331

INDIANA MICHIGAN POWER COMPANY'S EVALUATION

Subject: Deletion of the Power Range Neutron Flux High Negative Rate Trip Function

1.0 DESCRIPTION

This letter is a request by Indiana Michigan Power Company (I&M) to amend Facility Operating Licenses DPR-58 and DPR-74 for the Donald C. Cook Nuclear Plant (CNP) Units 1 and 2. The proposed changes would modify Technical Specifications (TS) to delete the power range neutron flux high negative rate trip. The proposed change will allow elimination of an unnecessary trip function and thereby reduce the potential for a transient.

2.0 PROPOSED CHANGE

By separate correspondence, Nuclear Regulatory Commission (NRC) has approved conversion of the CNP Current Technical Specifications (CTS) to the Improved Technical Specifications (ITS) specified in NUREG-1431. I&M intends to implement ITS no later than October 31, 2005; however, ITS have not yet been implemented. I&M has therefore provided copies of both the CTS and the ITS pages that are affected by this proposed amendment.

CTS Changes

In each of the following CTS tables, Functional Unit 4, Power Range, Neutron Flux, High Negative Rate, is deleted:

- CTS 2.2.1, Reactor Trip System Instrumentation Trip Setpoints, Table 2.2-1,
- CTS 3/4.3.1, Reactor Trip System Instrumentation, Table 3.3-1, and
- CTS 3/4.3.1, Reactor Trip System Instrumentation Surveillance Requirements, Table 4.3-1.

ITS Changes

ITS 3.3.1, Reactor Trip System Instrumentation, Table 3.3.1-1, Function 3.b is deleted.

In summary, the proposed change will modify TS to delete the power range neutron flux high negative rate trip. The proposed changes are consistent with the methodology presented in the Westinghouse Topical Report WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event" (Reference 1).

Changes to CTS Bases 2.2.1 and ITS Bases 3.3.1 are required to reflect deletion of the Power Range Neutron Flux – High Negative Rate trip (NFRT) function. These changes will be made in accordance with the Technical Specification Bases Control Program.

3.0 BACKGROUND

3.1 System Descriptions

The applicable system involved in the proposed amendment is the reactor protection system. The CNP Updated Final Safety Analysis Report (UFSAR), Section 7.2, Protective Systems, states that the protective systems consist of both the reactor protection system and the engineered safety features. All equipment from sensors to actuating devices is considered a part of that protective system. Design criteria for protection systems permit maximum effective use of process measurements both for control and protection functions, thus enhancing the capability to provide an adequate system to deal with the majority of common mode failures as well as to provide redundancy for critical control functions. This diversity in the design approach provides a protection system which monitors numerous system variables by different means.

The basic reactor operating philosophy is to define an allowable region of power, pressure, and coolant temperature conditions. This allowable range is defined by the primary tripping functions – the overpower delta-T trip, the overtemperature delta-T trip, and the nuclear overpower trip. The operating region below these trip settings is designed so that no combination of power, temperatures, and pressures could result in departure from nucleate boiling ratio (DNBR) less than the minimum DNBR for any credible operational transient when at power. Tripping functions in addition to those stated above are provided to back up the primary tripping functions for specific abnormal conditions.

UFSAR Section 7.2.4 discusses how the reactor protection system prevents departure from nucleate boiling (DNB). Plant variables affecting DNB are thermal power, reactor coolant system (RCS) flow, RCS temperature, RCS pressure, and core power distribution. Reactor trips for a high pressurizer pressure and for a fixed low pressurizer pressure are provided to limit the pressure range over which core protection depends on the overpower and overtemperature delta-T trips. Reactor trips on nuclear overpower and low RCS flow are provided for direct, immediate protection against rapid changes in these parameters. However, for all cases in which the calculated DNBR approaches a minimum, a reactor trip on overpower and/or overtemperature delta-T would also be actuated.

3.2 Reason for Requesting Amendment

The deletion of the NFRT function eliminates an unnecessary trip function and thereby reduces the potential for a transient, which could challenge safe plant operation due to spurious trip signals.

4.0 TECHNICAL ANALYSIS

The original design basis for the NFRT function was to mitigate the consequences of one or more dropped rod cluster control assemblies (RCCAs). The intent was that in the event of one or more dropped RCCAs, the reactor trip system would detect the rapidly decreasing neutron flux (i.e. high negative flux rate) due to the dropped RCCA(s) and would trip the reactor, thus ending the transient and assuring that DNB limits were maintained.

In 1982, an evaluation prepared by Westinghouse Electric Corporation and documented in WCAP-10297-P-A, "Dropped Rod Methodology for Negative Flux Rate Trip Plants," (Reference 2) determined that the NFRT function was only required when a dropped RCCA or RCCA bank exceeded a specific reactivity worth threshold value. Any dropped RCCA or RCCA bank which had a reactivity worth below the threshold value would not require a reactor trip to maintain DNB limits. An additional evaluation method, WCAP-11394-P-A (Reference 1), was developed by Westinghouse Electric Corporation in 1987, which determined that sufficient DNB margin existed for Westinghouse plant designs and fuel types without the NFRT function regardless of the reactivity worth of the dropped RCCA or RCCA bank, subject to a plant/cycle-specific analysis. The NRC subsequently reviewed and approved (Reference 3) the Westinghouse analysis method and results and concluded that the analysis contains an acceptable procedure for analyzing the dropped RCCA event for which no credit is taken for any direct reactor trip due to the dropped RCCA(s) or for automatic power reduction due to the dropped RCCA(s). Therefore, the NFRT function is not required to maintain existing DNB limits and may be eliminated.

The following provides an assessment of the proposed change with respect to other CNP safety analyses and evaluations.

Loss of Coolant Accident (LOCA) and LOCA-Related Evaluations

The NFRT function is not modeled in the LOCA analyses. The following LOCA-related analyses are not affected by the proposed changes: large and small break LOCA, reactor vessel and RCS loop LOCA blowdown forces, post-LOCA long term core cooling subcriticality, post-LOCA long term core cooling minimum flow, and RCS hot leg switchover to prevent boron precipitation. The proposed changes do not affect the normal plant operating parameters, accident mitigation capabilities important to a LOCA, the assumptions used in the LOCA-related accidents, or create conditions more limiting than those assumed in these analyses.

Non-LOCA Related Evaluation

The current non-LOCA safety analyses do not take credit for the NFRT function. Specifically, the dropped RCCA(s) analyses utilized for the current Unit 1 and Unit 2 cycles do not rely on actuation of the NFRT function to mitigate the consequences of the accident. These analyses were performed in accordance with the NRC approved methodology for the analysis of dropped RCCA(s) events provided in WCAP-11394-P-A. The analysis assumptions and confirmation that the DNB design basis is met are further confirmed as part of the reload safety analysis for each reactor core reload. The current reload safety analysis limits for CNP Unit 1 Cycle 20 and Unit 2 Cycle 15 confirm that DNB predicted for the dropped RCCA remains within safety analysis values. Therefore, the conclusion presented in the UFSAR, Chapter 14, that the DNB design basis is met with respect to non-LOCA related evaluations remains valid for the proposed changes which credit the application of WCAP-11394-P-A.

Mechanical Components and Systems Evaluation
Elimination of the NFRT function as described above does not affect the RCS component integrity or the ability of the RCS to perform its intended safety function. The proposed changes do not affect the integrity of plant systems or their ability to perform intended safety functions.

Containment Integrity Evaluation (Short Term / Long Term LOCA Case)
The NFRT function is not credited in the containment analyses. The proposed changes do not adversely affect the short term and long term LOCA mass and energy releases of the containment analyses. The proposed changes do not affect the normal plant operating parameters, system actuations, capabilities or assumptions important to the containment analyses, or create conditions more limiting than those assumed in these analyses. Therefore, the conclusions presented in the UFSAR remain valid with respect to the containment analyses.

Main Steam Line Break (MSLB) Mass and Energy Release Evaluation
The NFRT function is not credited in the UFSAR MSLB analyses. The proposed changes do not adversely affect the MSLB mass and energy releases, either inside or outside containment, and do not adversely affect the calculations for the steam mass release used as input to the radiological dose evaluation. The proposed changes do not affect the normal plant operating parameters, input assumptions, results or conclusions of the MSLB mass and energy release analyses, and steam release calculations. Also, conditions are not created which are more limiting than those enveloped by the current analyses and calculations. Therefore, the conclusions presented in the UFSAR remain valid with respect to MSLB mass and energy release rates and steam mass release calculations.

Emergency Operating Procedures (EOPs) Evaluation
Elimination of the NFRT function will not adversely affect the EOPs. Responding to dropped or misaligned RCCA events are covered by Abnormal Operating Procedures which instruct the operators to manually trip the reactor for multiple dropped RCCAs.

Safety Systems Allowable Values and Setpoints Evaluation
The NFRT function deletion does not change the current Allowable Value information for any other function shown in the TS, and does not change the current setpoint information for any other function shown in the Technical Requirements Manual (TRM). Therefore, since no credit for the NFRT function is taken in the safety analysis, the NFRT deletion has no impact on the plant safety functions.

Steam Generator Tube Rupture (SGTR) Evaluation
The NFRT function is not credited in the SGTR analyses. The proposed changes do not adversely affect the normal plant operating parameters, results or conclusions of the SGTR thermal and hydraulic analyses. Also, conditions are not created which are more limiting than

those enveloped by the current analyses for break flow and steam release. Therefore, the conclusions presented in the UFSAR remain valid with respect to the SGTR event.

Control Systems Evaluation
The proposed changes have no adverse impact on the control systems evaluation as documented in the Improved Thermal Design Procedure (Reference 4) and the Revised Thermal Design Procedure (Reference 5). The deletion of the NFRT function could increase plant availability because the proposed changes eliminate a potential source of inadvertent reactor trips.

For the past several fuel cycle designs, a dropped Rod Cluster Control Assembly (RCCA) analysis has been performed in accordance with the methodology described in WCAP-11394-P-A. Performance of the dropped RCCA analysis for future fuel cycle designs will be formalized in the CNP Nuclear Fuel administrative procedure for core designs. The NFRT function is not credited in the current cycle-specific dropped RCCA analysis, and the current analysis and limits conform to WCAP-11394-P-A.

5.0 REGULATORY SAFETY ANALYSIS

5.1 No Significant Hazards Consideration

Indiana Michigan Power Company (I&M) has evaluated whether a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability of occurrence or consequences of an accident previously evaluated?

   Response: No

   The removal of the power range neutron flux high negative rate trip function from technical specifications does not increase the probability or consequences of reactor core damage accidents resulting from dropped Rod Cluster Control Assembly (RCCA) events previously analyzed. The safety functions of other safety-related systems and components, which are related to mitigation of these events, have not been altered. All other Reactor Trip System and Engineered Safety Features Actuation Systems protection functions are not impacted by the elimination of the trip function. The dropped RCCA accident analysis does not rely on the negative flux rate trip to safely shut down the plant. The safety analysis of the plant is unaffected by the proposed change. Since the safety analysis is unaffected, the calculated radiological releases associated with the analysis are not affected.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   The proposed change does not adversely alter the design assumptions, conditions, or configuration of the facility or the manner in which the plant is operated. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed change. The proposed change does not challenge the performance or integrity of any safety-related systems or components. Nuclear Regulatory Commission (NRC)-approved Westinghouse Topical Report WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," dated January 1990 has demonstrated that the negative flux rate trip function can be eliminated.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No

   The margin of safety associated with the acceptance criteria of any accident is unchanged. It has been demonstrated that the negative flux rate trip function can be eliminated by the NRC-approved methodology described in WCAP-11394-P-A. Donald C. Cook Nuclear Plant cycle-specific analyses have confirmed that for a dropped RCCA(s) event, limits on departure from nucleate boiling are not exceeded by eliminating the negative flux rate trip. The proposed change will have no affect on the availability, operability, or performance of safety-related systems and components.

   Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Based on the above, I&M concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

10 CFR 50.36 (c) (2) (ii), stipulates that a technical specification limiting condition for operation (LCO) must be established for each item meeting one or more of the following criteria:

1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.
2. A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
3. A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
4. A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Since the power range neutron flux high negative rate trip function is not credited in safety analysis, the function is not considered an LCO in accordance with 10 CFR 50.36. That is, it does not meet any of the four criteria of 10 CFR 50.36, and therefore the function does not warrant inclusion in the technical specifications as an LCO.

In conclusion, based on the considerations discussed above, (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 or safety of the public.

6.0 ENVIRONMENTAL CONSIDERATIONS

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. Westinghouse Topical Report, WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," dated January 1990.
2. Westinghouse Topical Report WCAP-10297-P-A, "Dropped Rod Methodology for Negative Flux Rate Trip Plants," dated June 1983.
3. Letter from A. C. Thadani (NRC) to R. A. Newton (Westinghouse Owners Group), "Acceptance for Referencing of Licensing Topical Reports WCAP-11394(P) and WCAP-11395(NP), 'Methodology for the Analysis of the Dropped Rod Event'," dated October 23, 1989.
4. Westinghouse Topical Report WCAP-12568, Revision 1, "Westinghouse Improved Thermal Design Procedure, Instrument Uncertainty Methodology for American Electric Power, Donald C. Cook Unit 1," dated August 1993.
5. Westinghouse Topical Report WCAP-12576, Revision 1, "Westinghouse Revised Thermal Design Procedure, Instrument Uncertainty Methodology for American Electric Power, Donald C. Cook Unit 2," dated August 1993.

8.0 PRECEDENT

The NRC has approved similar submittals at plants deleting the power range neutron flux negative rate trip.

Seabrook	Accession No. ML032310339
Braidwood / Byron	Accession No. ML011410291
Watts Bar	Accession No. ML020780104

Attachment 1A to AEP:NRC:5331

DONALD C. COOK NUCLEAR PLANT UNIT 1 CURRENT TECHNICAL SPECIFICATION PAGES MARKED TO SHOW CHANGES

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3/4 3-3

3/4 3-12

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1.	Manual Reactor Trip	Not Applicable	Not Applicable
2.	Power Range, Neutron Flux	Low Setpoint - less than or equal to 25% of RATED THERMAL POWER High Setpoint - less than or equal to 109% of RATED THERMAL POWER	Low Setpoint - less than or equal to 26% of RATED THERMAL POWER High Setpoint - less than or equal to 110% of RATED THERMAL POWER
3.	Power Range, Neutron Flux, High Positive Rate	Less than or equal to 5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds	Less than or equal to 5.5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds
4.	~~Power Range, Neutron Flux, High Negative Rate~~DELETED	~~Less than or equal to 5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds~~	~~Less than or equal to 5.5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds~~
5.	Intermediate Range, Neutron Flux	Less than or equal to 25% of RATED THERMAL POWER	Less than or equal to 30% of RATED THERMAL POWER
6.	Source Range, Neutron Flux	Less than or equal to 10^5 counts per second	Less than or equal to 1.3×10^5 counts per second
7.	Overtemperature Delta T	See Note 1	See Note 3
8.	Overpower Delta T	See Note 2	See Note 4
9.	Pressurizer Pressure -- Low	Greater than or equal to 1875 psig	Greater than or equal to 1865 psig
10.	Pressurizer Pressure -- High	Less than or equal to 2385 psig	Less than or equal to 2395 psig
11.	Pressurizer Water Level -- High	Less than or equal to 92% of instrument span	Less than or equal to 93% of instrument span
12.	Loss of Flow	Greater than or equal to 90% of design flow per loop*	Greater than or equal to 89.1% of design flow per loop*

*Design flow is 1/4 Reactor Coolant System total flow rate from Table 3.2-1.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

	FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1.	Manual Reactor Trip	2	1	2	1, 2 and *	12
2.	Power Range, Neutron Flux	4	2	3	1, 2 and *	2
3.	Power Range, Neutron Flux, High Positive Rate	4	2	3	1, 2	2
4.	~~Power Range, Neutron Flux, High Negative Rate~~DELETED	4	~~2~~	~~3~~	~~1, 2~~	~~2~~
5.	Intermediate Range, Neutron Flux	2	1	2	1, 2 and *	3
6.	Source Range, Neutron Flux					
	A. Startup	2	1	2	2## and *	4
	B. Shutdown	2	0	1	3, 4 and 5	5
7.	Overtemperature ΔT Four Loop Operation	4	2	3	1, 2	6
8.	Overpower ΔT Four Loop Operation	4	2	3	1, 2	6

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODE IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip				
A. Shunt Trip Function	N.A.	N.A.	S/U(1)(10)	1, 2, 3*, 4*, 5*
B. Undervoltage Trip Function	N.A.	N.A.	S/U(1)(10)	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux	S	D(2), M(3), and Q(6)	Q and S/U(1)	1, 2 and *
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(6)	Q	1, 2
4. ~~Power Range, Neutron Flux, High Negative Rate~~DELETED	~~N.A.~~	~~R(6)~~	~~Q~~	~~1, 2~~
5. Intermediate Range, Neutron Flux	S	R(6)	S/U(17)	1, 2, and *
6. Source Range, Neutron Flux	S	R(6,14)	M(14) and S/U(1)	2(7), 3(7), 4 and 5
7. Overtemperature delta T	S	R	SA	1, 2
8. Overpower delta T	S	R	SA	1, 2
9. Pressurizer Pressure -- Low	S	R	SA	1, 2
10. Pressurizer Pressure -- High	S	R	SA	1, 2
11. Pressurizer Water Level -- High	S	R	SA	1, 2
12. Loss of Flow-Single Loop	S	R	SA	1

Attachment 1B to AEP:NRC:5331

DONALD C. COOK NUCLEAR PLANT UNIT 2 CURRENT TECHNICAL SPECIFICATION PAGES MARKED TO SHOW CHANGES

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3/4 3-11

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1.	Manual Reactor Trip	Not Applicable	Not Applicable
2.	Power Range, Neutron Flux	Low Setpoint - Less than or equal to 25% of RATED THERMAL POWER	Low Setpoint - Less than or equal to 26% of RATED THERMAL POWER
		High Setpoint - Less than or equal to 109% of RATED THERMAL POWER	High Setpoint - Less than or equal to 110% of RATED THERMAL POWER
3.	Power Range, Neutron Flux, High Positive Rate	Less than or equal to 5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds	Less than or equal to 5.5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds
4.	~~Power Range, Neutron Flux, High Negative Rate~~DELETED	~~Less than or equal to 5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds~~	~~Less than or equal to 5.5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds~~
5.	Intermediate Range, Neutron Flux	Less than or equal to 25% of RATED THERMAL POWER	Less than or equal to 30% of RATED THERMAL POWER
6.	Source Range, Neutron Flux	Less than or equal to 10^5 counts per second	Less than or equal to 1.3×10^5 counts per second
7.	Overtemperature Delta T	See Note 1	See Note 3
8.	Overpower Delta T	See Note 2	See Note 4
9.	Pressurizer Pressure -- Low	Greater than or equal to 1950 psig	Greater than or equal to 1940 psig
10.	Pressurizer Pressure -- High	Less than or equal to 2385 psig	Less than or equal to 2395 psig
11.	Pressurizer Water Level -- High	Less than or equal to 92% of instrument span	Less than or equal to 93% of instrument span
12.	Loss of Flow	Greater than or equal to 90% of design flow per loop*	Greater than or equal to 89.1% of design flow per loop*

*Design flow is 91,600 gpm per loop.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

	FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1.	Manual Reactor Trip	2	1	2	1, 2 and *	12
2.	Power Range, Neutron Flux	4	2	3	1, 2 and *	2
3.	Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2
4.	~~Power Range, Neutron Flux High Negative Rate~~DELETED	4	~~2~~	~~3~~	~~1, 2~~	~~2~~
5.	Intermediate Range, Neutron Flux	2	1	2	1, 2 and *	3
6.	Source Range, Neutron Flux					
	A. Startup	2	1	2	2## and *	4
	B Shutdown	2	0	1	3, 4 and 5	5
7.	Overtemperature ΔT Four Loop Operation	4	2	3	1, 2	6
8.	Overpower ΔT Four Loop Operation	4	2	3	1, 2	6

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip				
A. Shunt Trip Function	N.A.	N.A.	S/U(1)(10)	1, 2, 3*, 4*, 5*
B. Undervoltage Trip Function	N.A.	N.A.	S/U(1)(10)	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux	S	D(2), M(3) and Q(6)	M and S/U(1)	1, 2 and *
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(6)	M	1, 2
4. ~~Power Range, Neutron Flux, High Negative Rate~~DELETED	~~N.A.~~	~~R(6)~~	~~M~~	~~1, 2~~
5. Intermediate Range, Neutron Flux	S	R(6,8)	S/U(1)	1, 2, and *
6. Source Range, Neutron Flux	S	R(6,14)	M(14) and S/U(1)	2(7), 3(7), 4 and 5
7. Overtemperature ΔT	S	R(9)	M	1, 2
8. Overpower ΔT	S	R(9)	M	1, 2
9. Pressurizer Pressure -- Low	S	R	M	1, 2
10. Pressurizer Pressure -- High	S	R	M	1, 2
11. Pressurizer Water Level -- High	S	R	M	1, 2
12. Loss of Flow-Single Loop	S	R(8)	M	1

Attachment 2A to AEP:NRC:5331

DONALD C. COOK NUCLEAR PLANT UNIT 1 CURRENT TECHNICAL SPECIFICATION PAGES WITH THE PROPOSED CHANGES INCORPORATED

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3/4 3-12

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1.	Manual Reactor Trip	Not Applicable	Not Applicable
2.	Power Range, Neutron Flux	Low Setpoint - less than or equal to 25% of RATED THERMAL POWER High Setpoint - less than or equal to 109% of RATED THERMAL POWER	Low Setpoint - less than or equal to 26% of RATED THERMAL POWER High Setpoint - less than or equal to 110% of RATED THERMAL POWER
3.	Power Range, Neutron Flux, High Positive Rate	Less than or equal to 5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds	Less than or equal to 5.5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds
4.	DELETED		
5.	Intermediate Range, Neutron Flux	Less than or equal to 25% of RATED THERMAL POWER	Less than or equal to 30% of RATED THERMAL POWER
6.	Source Range, Neutron Flux	Less than or equal to 10^5 counts per second	Less than or equal to 1.3×10^5 counts per second
7.	Overtemperature Delta T	See Note 1	See Note 3
8.	Overpower Delta T	See Note 2	See Note 4
9.	Pressurizer Pressure -- Low	Greater than or equal to 1875 psig	Greater than or equal to 1865 psig
10.	Pressurizer Pressure -- High	Less than or equal to 2385 psig	Less than or equal to 2395 psig
11.	Pressurizer Water Level -- High	Less than or equal to 92% of instrument span	Less than or equal to 93% of instrument span
12.	Loss of Flow	Greater than or equal to 90% of design flow per loop*	Greater than or equal to 89.1% of design flow per loop*

*Design flow is 1/4 Reactor Coolant System total flow rate from Table 3.2-1.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

	FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1.	Manual Reactor Trip	2	1	2	1, 2 and *	12
2.	Power Range, Neutron Flux	4	2	3	1, 2 and *	2
3.	Power Range, Neutron Flux, High Positive Rate	4	2	3	1, 2	2
4.	DELETED					
5.	Intermediate Range, Neutron Flux	2	1	2	1, 2 and *	3
6.	Source Range, Neutron Flux					
	A. Startup	2	1	2	$2^{\#\#}$ and *	4
	B. Shutdown	2	0	1	3, 4 and 5	5
7.	Overtemperature ΔT Four Loop Operation	4	2	3	1, 2	6
8.	Overpower ΔT Four Loop Operation	4	2	3	1, 2	6

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODE IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip				
A. Shunt Trip Function	N.A.	N.A.	S/U(1)(10)	1, 2, 3*, 4*, 5*
B. Undervoltage Trip Function	N.A.	N.A.	S/U(1)(10)	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux	S	D(2), M(3), and Q(6)	Q and S/U(1)	1, 2 and *
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(6)	Q	1, 2
4. DELETED				
5. Intermediate Range, Neutron Flux	S	R(6)	S/U(17)	1, 2, and *
6. Source Range, Neutron Flux	S	R(6,14)	M(14) and S/U(1)	2(7), 3(7), 4 and 5
7. Overtemperature delta T	S	R	SA	1, 2
8. Overpower delta T	S	R	SA	1, 2
9. Pressurizer Pressure -- Low	S	R	SA	1, 2
10. Pressurizer Pressure -- High	S	R	SA	1, 2
11. Pressurizer Water Level -- High	S	R	SA	1, 2
12. Loss of Flow-Single Loop	S	R	SA	1

Attachment 2B to AEP:NRC:5331

DONALD C. COOK NUCLEAR PLANT UNIT 2 CURRENT TECHNICAL SPECIFICATION PAGES WITH THE PROPOSED CHANGES INCORPORATED

2-5

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3/4 3-11

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

	FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1.	Manual Reactor Trip	Not Applicable	Not Applicable
2.	Power Range, Neutron Flux	Low Setpoint - Less than or equal to 25% of RATED THERMAL POWER	Low Setpoint - Less than or equal to 26% of RATED THERMAL POWER
		High Setpoint - Less than or equal to 109% of RATED THERMAL POWER	High Setpoint - Less than or equal to 110% of RATED THERMAL POWER
3.	Power Range, Neutron Flux, High Positive Rate	Less than or equal to 5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds	Less than or equal to 5.5% of RATED THERMAL POWER with a time constant greater than or equal to 2 seconds
4.	DELETED		
5.	Intermediate Range, Neutron Flux	Less than or equal to 25% of RATED THERMAL POWER	Less than or equal to 30% of RATED THERMAL POWER
6.	Source Range, Neutron Flux	Less than or equal to 10^5 counts per second	Less than or equal to 1.3×10^5 counts per second
7.	Overtemperature Delta T	See Note 1	See Note 3
8.	Overpower Delta T	See Note 2	See Note 4
9.	Pressurizer Pressure -- Low	Greater than or equal to 1950 psig	Greater than or equal to 1940 psig
10.	Pressurizer Pressure -- High	Less than or equal to 2385 psig	Less than or equal to 2395 psig
11.	Pressurizer Water Level -- High	Less than or equal to 92% of instrument span	Less than or equal to 93% of instrument span
12.	Loss of Flow	Greater than or equal to 90% of design flow per loop*	Greater than or equal to 89.1% of design flow per loop*

*Design flow is 91,600 gpm per loop.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

	FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1.	Manual Reactor Trip	2	1	2	1, 2 and *	12
2.	Power Range, Neutron Flux	4	2	3	1, 2 and *	2
3.	Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2
4.	DELETED					
5.	Intermediate Range, Neutron Flux	2	1	2	1, 2 and *	3
6.	Source Range, Neutron Flux					
	A. Startup	2	1	2	2## and *	4
	B Shutdown	2	0	1	3, 4 and 5	5
7.	Overtemperature ΔT Four Loop Operation	4	2	3	1, 2	6
8.	Overpower ΔT Four Loop Operation	4	2	3	1, 2	6

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip				
A. Shunt Trip Function	N.A.	N.A.	S/U(1)(10)	1, 2, 3*, 4*, 5*
B. Undervoltage Trip Function	N.A.	N.A.	S/U(1)(10)	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux	S	D(2), M(3) and Q(6)	M and S/U(1)	1, 2 and *
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(6)	M	1, 2
4. DELETED				
5. Intermediate Range, Neutron Flux	S	R(6,8)	S/U(1)	1, 2, and *
6. Source Range, Neutron Flux	S	R(6,14)	M(14) and S/U(1)	2(7), 3(7), 4 and 5
7. Overtemperature ΔT	S	R(9)	M	1, 2
8. Overpower ΔT	S	R(9)	M	1, 2
9. Pressurizer Pressure -- Low	S	R	M	1, 2
10. Pressurizer Pressure -- High	S	R	M	1, 2
11. Pressurizer Water Level -- High	S	R	M	1, 2
12. Loss of Flow-Single Loop	S	R(8)	M	1

Attachment 3A to AEP:NRC:5331

DONALD C. COOK NUCLEAR PLANT UNIT 1 IMPROVED TECHNICAL SPECIFICATION PAGES MARKED TO SHOW CHANGES

3.3.1-11

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.17	NA
	$3^{(a)}$, $4^{(a)}$, $5^{(a)}$	2	B	SR 3.3.1.17	NA
2. Power Range Neutron Flux					
a. High	1,2	4	C	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.8 SR 3.3.1.9 SR 3.3.1.19	≤ 110% RTP
b. Low	$1^{(b)}$,2	4	D	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.9 SR 3.3.1.19	≤ 26% RTP
3. Power Range Neutron Flux ~~Rate~~					
~~a.~~ High Positive Rate	1,2	4	D	SR 3.3.1.8 SR 3.3.1.14	≤ 5.5% RTP with time constant ≥ 2 sec
~~b. High Negative Rate~~	~~1,2~~	4	~~D~~	~~SR 3.3.1.8~~ ~~SR 3.3.1.14~~	~~≤ 5.5% RTP with time constant ≥ 2 sec~~
4. Intermediate Range Neutron Flux	$1^{(b)}$, $2^{(c)}$	2	E, F	SR 3.3.1.1 SR 3.3.1.11 SR 3.3.1.14	≤ 30% RTP
5. Source Range Neutron Flux	$2^{(d)}$	2	G, H	SR 3.3.1.1 SR 3.3.1.11 SR 3.3.1.14	≤ 1.3E5 cps
	$3^{(a)}$, $4^{(a)}$, $5^{(a)}$	2	H, I	SR 3.3.1.1 SR 3.3.1.11 SR 3.3.1.14	≤ 1.3E5 cps

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

(b) Below the P-10 (Power Range Neutron Flux) interlock.

(c) Above the P-6 (Intermediate Range Neutron Flux) interlock.

(d) Below the P-6 (Intermediate Range Neutron Flux) interlock.

Attachment 3B to AEP:NRC:5331

DONALD C. COOK NUCLEAR PLANT UNIT 2 IMPROVED TECHNICAL SPECIFICATION PAGES MARKED TO SHOW CHANGES

3.3.1-11

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.17	NA
	$3^{(a)}, 4^{(a)}, 5^{(a)}$	2	B	SR 3.3.1.17	NA
2. Power Range Neutron Flux					
a. High	1,2	4	C	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.8 SR 3.3.1.9 SR 3.3.1.19	≤ 110% RTP
b. Low	$1^{(b)}, 2$	4	D	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.9 SR 3.3.1.19	≤ 26% RTP
3. Power Range Neutron Flux ~~Rate~~					
~~a.~~ High Positive Rate	1,2	4	D	SR 3.3.1.8 SR 3.3.1.14	≤ 5.5% RTP with time constant ≥ 2 sec
~~b. High Negative Rate~~	~~1,2~~	4	~~D~~	~~SR 3.3.1.8~~ ~~SR 3.3.1.14~~	~~≤ 5.5% RTP with time constant ≥ 2 sec~~
4. Intermediate Range Neutron Flux	$1^{(b)}, 2^{(c)}$	2	E, F	SR 3.3.1.1 SR 3.3.1.11 SR 3.3.1.14	≤ 30% RTP
5. Source Range Neutron Flux	$2^{(d)}$	2	G, H	SR 3.3.1.1 SR 3.3.1.11 SR 3.3.1.14	≤ 1.3E5 cps
	$3^{(a)}, 4^{(a)}, 5^{(a)}$	2	H, I	SR 3.3.1.1 SR 3.3.1.11 SR 3.3.1.14	≤ 1.3E5 cps

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

(b) Below the P-10 (Power Range Neutron Flux) interlock.

(c) Above the P-6 (Intermediate Range Neutron Flux) interlock.

(d) Below the P-6 (Intermediate Range Neutron Flux) interlock.

Attachment 4A to AEP:NRC:5331

DONALD C. COOK NUCLEAR PLANT UNIT 1 IMPROVED TECHNICAL SPECIFICATION PAGES WITH THE PROPOSED CHANGES INCORPORATED

3.3.1-11

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.17	NA
	$3^{(a)}, 4^{(a)}, 5^{(a)}$	2	B	SR 3.3.1.17	NA
2. Power Range Neutron Flux					
a. High	1,2	4	C	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.8 SR 3.3.1.9 SR 3.3.1.19	≤ 110% RTP
b. Low	$1^{(b)}, 2$	4	D	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.9 SR 3.3.1.19	≤ 26% RTP
3. Power Range Neutron Flux - High Positive Rate	1,2	4	D	SR 3.3.1.8 SR 3.3.1.14	≤ 5.5% RTP with time constant ≥ 2 sec
4. Intermediate Range Neutron Flux	$1^{(b)}, 2^{(c)}$	2	E, F	SR 3.3.1.1 SR 3.3.1.11 SR 3.3.1.14	≤ 30% RTP
5. Source Range Neutron Flux	$2^{(d)}$	2	G, H	SR 3.3.1.1 SR 3.3.1.11 SR 3.3.1.14	≤ 1.3E5 cps
	$3^{(a)}, 4^{(a)}, 5^{(a)}$	2	H, I	SR 3.3.1.1 SR 3.3.1.11 SR 3.3.1.14	≤ 1.3E5 cps

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

(b) Below the P-10 (Power Range Neutron Flux) interlock.

(c) Above the P-6 (Intermediate Range Neutron Flux) interlock.

(d) Below the P-6 (Intermediate Range Neutron Flux) interlock.

Attachment 4B to AEP:NRC:5331

DONALD C. COOK NUCLEAR PLANT UNIT 2 IMPROVED TECHNICAL SPECIFICATION PAGES WITH THE PROPOSED CHANGES INCORPORATED

3.3.1-11

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

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1.	Manual Reactor Trip	1,2	2	B	SR 3.3.1.17	NA
		3[(a)], 4[(a)], 5[(a)]	2	B	SR 3.3.1.17	NA
2.	Power Range Neutron Flux					
	a. High	1,2	4	C	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.8 SR 3.3.1.9 SR 3.3.1.19	≤ 110% RTP
	b. Low	1[(b)],2	4	D	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.9 SR 3.3.1.19	≤ 26% RTP
3.	Power Range Neutron Flux - High Positive Rate	1,2	4	D	SR 3.3.1.8 SR 3.3.1.14	≤ 5.5% RTP with time constant ≥ 2 sec
4.	Intermediate Range Neutron Flux	1[(b)], 2[(c)]	2	E, F	SR 3.3.1.1 SR 3.3.1.11 SR 3.3.1.14	≤ 30% RTP
5.	Source Range Neutron Flux	2[(d)]	2	G, H	SR 3.3.1.1 SR 3.3.1.11 SR 3.3.1.14	≤ 1.3E5 cps
		3[(a)], 4[(a)], 5[(a)]	2	H, I	SR 3.3.1.1 SR 3.3.1.11 SR 3.3.1.14	≤ 1.3E5 cps

(a) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

(b) Below the P-10 (Power Range Neutron Flux) interlock.

(c) Above the P-6 (Intermediate Range Neutron Flux) interlock.

(d) Below the P-6 (Intermediate Range Neutron Flux) interlock.

ATTACHMENT 5 TO AEP:NRC:5331

REGULATORY COMMITMENTS

The following table identifies those actions committed to by Indiana Michigan Power Company (I&M) in this document. Any other actions discussed in this submittal represent intended or planned actions by I&M. They are described to the Nuclear Regulatory Commission (NRC) for the NRC's information and are not regulatory commitments.

Commitment	Date
For the past several fuel cycle designs, a dropped Rod Cluster Control Assembly (RCCA) analysis has been performed in accordance with the methodology described in WCAP-11394-P-A. Performance of the dropped RCCA analysis for future fuel cycle designs will be formalized in the CNP Nuclear Fuel administrative procedure for core designs.	Prior to the start of each fuel cycle.