May 23, 2003

Mr. A. Christopher Bakken III, Senior Vice President
and Chief Nuclear Officer
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS (TAC NOS. MB6324 AND MB6325)

Dear Mr. Bakken:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 277 to Facility Operating License No. DPR-58 and Amendment No. 260 to Facility Operating License No. DPR-74 for the Donald C. Cook Nuclear Plant, Units 1 and 2. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated August 30, 2002, as supplemented by letters dated February 27, April 7, April 29, and May 2, 2003.

The amendments revise the reactor trip system and engineered safety features actuation system (ESFAS) surveillance requirements, increasing selected surveillance intervals for analog channels, logic cabinets, and reactor trip breakers and increasing the completion time and bypass time for the reactor trip breakers. These changes were proposed in accordance with WCAP-15376-P, Revision 0, “Risk-Informed Assessment of the RTS and ESFAS surveillance test intervals and reactor trip breaker test and completion times,” which was accepted by the Nuclear Regulatory Commission (NRC) staff for referencing as documented in a letter dated December 20, 2002, from R. H. Ruland, NRC, to R. H. Bryan, Westinghouse Owners Group.

As noted above, you submitted four supplements to your original application. These supplements removed proposed changes that were not within the scope of the accepted version of WCAP-15376-P; corrected errors in proposed TS requirements; and provided additional technical justifications to address NRC staff questions. While we recognize that the proposed amendments involved complex technical and regulatory issues, complete and accurate amendment applications ensure technically accurate and timely reviews by the NRC staff. Amendment applications which require several supplements present unnecessary challenges to the completion of technically accurate, effective, and efficient reviews by the NRC staff.

A copy of our related safety evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures:
1. Amendment No. 277 to DPR-58
2. Amendment No. 260 to DPR-74
3. Safety Evaluation

cc w/encls: See next page

A copy of our related safety evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

John F. Stang, Senior Project Manager, Section 1
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Docket Nos. 50-315 and 50-316

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ADAMS ACCESSION NUMBER: ML031320614 *See attached concurrence sheet

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INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 277
License No. DPR-58

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

   A. The application for amendment by Indiana Michigan Power Company (the licensee) application dated August 30, 2002 as supplemented by letters dated February 27, April 7, April 29, and May 2, 2003, 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-58 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 277, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by L. Marsh for/

L. Raghavan, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: May 23, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 277

TO FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
1-9	1-9
3/4 3-5	3/4 3-5
3/4 3-8	3/4 3-8
3/4 3-12	3/4 3-12
3/4 3-13	3/4 3-13
3/4 3-14	3/4 3-14
3/4 3-31	3/4 3-31
3/4 3-32	3/4 3-32
3/4 3-33	3/4 3-33
3/4 3-33a	3/4 3-33a
3/4 3-33b	3/4 3-33b
3/4 3-34	3/4 3-34

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 260
License No. DPR-74

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

   A. The application for amendment by Indiana Michigan Power Company (the licensee) application dated August 30, 2002 as supplemented by letters dated February 27, April 7, April 29, and May 2, 2003, 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:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 260, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by L. Marsh for/

L. Raghavan, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: May 23, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 260

FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
1-10	1-10
3/4 3-4	3/4 3-4
3/4 3-7	3/4 3-7
3/4 3-11	3/4 3-11
3/4 3-12	3/4 3-12
3/4 3-13	3/4 3-13
3/4 3-30	3/4 3-30
3/4 3-31	3/4 3-31
3/4 3-32	3/4 3-32
3/4 3-33	3/4 3-33

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 277 TO FACILITY OPERATING LICENSE NO. DPR-58

AND AMENDMENT NO. 260 TO FACILITY OPERATING LICENSE NO. DPR-74

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-315 AND 50-316

1.0 INTRODUCTION

By application dated August 30, 2002 as supplemented by letters dated February 27, April 7, April 29, and May 2, 2003, the Indiana Michigan Power Company (the licensee) requested amendments to the Technical Specifications (TSs) for the Donald C. Cook Nuclear Plant, Units 1 and 2. The proposed amendments would revise the reactor trip system (RTS) surveillance requirement, TS 3/4.3.1, and engineered safety features actuation system (ESFAS) surveillance requirement, TS 3/4.3.2, by increasing the channel operational test surveillance intervals for analog channels, logic cabinets, and reactor trip breakers. Additionally, the proposed amendments would revise the RTS surveillance requirement, TS 3/4.3.1 and ESFAS surveillance requirement, TS 3/4.3.2, increasing the completion time and bypass time for the reactor trip breakers. These changes were proposed in accordance with WCAP-15376-P, Revision 0, “Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times,” and the Nuclear Regulatory Commission (NRC) staff’s approved Technical Specification Task Force (TSTF) Traveler TSTF-411, Revision 1, “Surveillance Test Interval Extension for Components of the Reactor Protection System.”

The supplemental letters contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original *Federal Register* notice.

2.0 REGULATORY EVALUATION

On February 6, 1987, the Commission noticed and issued in the Federal Register (52 FR 3788), guidelines for improving the content and quality of nuclear power plant TS, “Interim Policy Statement on Technical Specification Improvements for Nuclear Power Reactors.” During the period from 1989 to 1992, utility owners groups and the staff developed improved Standard Technical Specifications (STS) that would establish models based on the Commission’s Interim Policy Statement for each major reactor type.

Improved STS were developed based on the criteria in the Commission’s Interim Policy Statement. In September 1992, the Commission issued Revision 0 of the improved STS as

NUREGs 1430-1434. D. C. Cook, Units 1 and 2 use pressurized water nuclear steam supply systems designed by Westinghouse Electric Company, LLC. Improved STS for plants, such as D. C. Cook, Units 1 and 2, were published in NUREG-1431, "Standard Technical Specifications Westinghouse Plants, Revision 0."

On July 22, 1993, the "Final Commission Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" (58 FR 39132), was published. This was subsequently codified by changes to Section 36 of Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.36) (60 FR 36953).

10 CFR 50.36 provides regulatory requirements related to the content of Technical Specifications. Specifically, 10 CFR 50.36(c)(2)(ii) establishes that a limiting condition for operation (LCO) is required to be included in TSs 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 or presents a challenge to the integrity of a fission product barrier; or

4) A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Review of any proposed generic changes to the STS NUREGs is accomplished by a multi-stage process designed to ensure that the STS NUREGs: remain internally consistent; maintain coherence between each STS NUREG; and incorporate the knowledge and operating experience of the industry and the NRC.

Changes to the STS NUREGs, which are potentially applicable to multiple plants, are proposed to the NRC staff by the Nuclear Energy Institute (NEI) sponsored TSTF via publicly available submittals. The TSTF includes representatives from the four U.S. commercial nuclear power plant owner's groups and NEI. The NRC staff reviews the proposed changes (referred to as TSTFs) to the STS NUREGs and will accept, modify, or reject each proposed change. Once a proposed TSTF change has been accepted, it is considered part of the applicable STS NUREG. After NRC staff acceptance, licensees may propose to incorporate a TSTF change into their plant-specific TSs by way of the license amendment process described in 10 CFR 50.90.

The established STS NUREG revision process facilitates licensees adopting NRC-accepted changes to the STS NUREGs for their plant-specific TSs. This process is intended to streamline the license amendment review process for these types of proposed amendments in order to increase NRC efficiency and reduce unnecessary regulatory burden. The NRC role in maintaining plant safety is achieved by the technical review of proposed changes to the STS NUREGs as well as plant specific license amendment applications to adopt NRC-accepted changes. Licensees are encouraged to modify their TSs, to the extent practical and consistent with their design and licensing basis, to be consistent with the STS.

By letter dated November 8, 2000, as supplemented by letters dated June 8, June 25, and September 28, 2001, and January 8, 2002, the Westinghouse Owners Group submitted topical report WCAP-15376-P, Rev. 0, “Risk-informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times” to the NRC for review and approval. WCAP-15376-P provided technical evaluation and analysis of changes to the TSs for the RTS and ESFAS. The proposed changes include increasing the completion and bypass times for the reactor trip breakers. Additionally, the surveillance test intervals are increased for the reactor trip breakers, master relays, logic cabinets, and analog channels. The proposed changes were incorporated into proposed TSTF-411, Rev. 0, “Surveillance Test Interval Extension for Components of the Reactor Protection System.” TSTF-411, Revision 1, was submitted to the NRC staff by letter dated August 9, 2001.

On December 20, 2002, the NRC staff documented its acceptance of WCAP-15376-P, Rev. 0 and TSTF-411, Rev. 1, in a letter from W. R. Ruland, NRC, to R. H. Bryan, Westinghouse Owners Group. This letter noted that the NRC staff does not intend to repeat its review of matters described in WCAP-15376-P, Rev. 0, and found acceptable, when the report appears as a reference in license applications, except to ensure that the material presented applies to the specific plant involved. The safety evaluation enclosed with the December 20, 2002, acceptance letter, defines the basis for acceptance of WCAP-15376-P, Rev. 0, as well as specific conditions and limitations.

3.0 TECHNICAL EVALUATION

The NRC staff reviewed the licensee’s application, as supplemented, related to the implementation of WCAP-15376, Rev. 0 and TSTF-411 Rev. 1. The NRC staff‘s review followed the guidance provided in the NRC staff’s acceptance letter for WCAP-15376-P dated December 20, 2002. Included in this review was verification that the changes proposed, as adapted from the NUREG-1431 format, were bounded by the December 20, 2002, acceptance letter and enclosed safety evaluation. In addition, the NRC staff verified that the licensee had adequately addressed the conditions and limitations delineated in the enclosed safety evaluation.

The December 20, 2002 acceptance letter for WCAP-15376 noted that this topical report was built on the foundation established by WCAP-10271-P, “Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System,” and WCAP-14333, “Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times.” The NRC staff’s review of the licensee’s application, as supplemented, verified that the applicable implementation requirements associated with the NRC staff acceptance of WCAPs-10271 and 14333 were also adequately addressed by the licensee.

The licensee made several regulatory commitments that address these conditions and limitations and other implementation items. These regulatory commitments included the following items which will be completed prior to the implementation of the amendments:

A. Implement administrative controls in the configuration risk management program to include the following restrictions when a reactor trip beaker and/or logic cabinet is removed from service:

   Activities that degrade the availability of auxiliary feedwater, reactor coolant system pressure relief, anticipated transient without scram mitigating system actuation circuitry, or turbine trip should not be scheduled when a reactor trip breaker is out of service;

   Activities that could degrade the operable train of the reactor protection system including master relays, slave relays, and analog channels should not be scheduled concurrently with the out-of-service train; and

   Activities on electrical support systems for auxiliary feedwater, reactor coolant system pressure relief, anticipated transient without scram mitigating system actuation circuitry, or turbine trip should not be scheduled during reactor trip breaker maintenance;

B. Establish administrative controls to ensure any future digital upgrades to the reactor protection system and/or engineered safety features actuation system are evaluated to ensure that the generic applicability of WCAP-15376-P is not affected;

C. Implement procedures to document, during Operations review of conditions adverse to quality, plausible common causes for equipment failures, and to initiate testing/inspection if necessary to determine operability of affected licensing basis equipment;

D. Establish controls prohibiting routine surveillance procedures from testing reactor trip system analog channels in bypass through the use of lifted leads or jumpers;

E. Establish a program to monitor and review as-found and as-left data for the power range nuclear instrument channels for a one year period, starting at implementation, to verify that the observed setpoint drift remains within the existing allowance contained in the instrument setpoint calculation.

The NRC staff noted, during its review, that the existing D. C. Cook Unit 1 and 2 TSs do not include specific functional units or separate TS LCOs for all RTS and ESFAS master and slave relays. The licensee stated in their April 7, 2003, response to the NRC staff's request for additional information, that the periodic testing of these master and slave relays is procedurally required to be performed to verify system operability. The licensee provided a regulatory commitment, in a letter dated April 29, 2003, to include RTS and ESFAS master and slave relay

testing specified in NUREG-1431 in its scheduled conversion[1] to the current STS. The licensee has scheduled to submit, in the first quarter 2004, an application for amendment converting to the current STS. The NRC staff has concluded that this is acceptable.

The NRC staff also noted, during its review, that the existing D. C. Cook Unit 1 and 2 TSs do not include the definition of a CHANNEL OPERATIONAL TEST. NUREG-1431 establishes the definition of a "CHANNEL OPERATIONAL TEST." This test is defined, in part, as, " ... the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY." The term "CHANNEL FUNCTIONAL TEST" is used in the D. C. Cook Unit 1 and 2 TSs to require a test which is analogous to the NUREG-1431 CHANNEL OPERATIONAL TEST. The NRC staff has concluded that this is acceptable.

The licensee's application as supplemented also included a small number of administrative and editorial changes. The administrative changes were needed to address plant-specific TS differences from the presentation of information in the current STS. These differences include definitions and the physical layout of information in the TSs. Editorial changes modifying punctuation were proposed to more clearly show the intended ACTIONS for two selected RTS functional units. These editorial changes do not alter the intent or applicability of the current TSs.

Table 3-1 provides a summary of the technical specification changes and the NRC staff's review of these changes.

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[1]Letter dated November 1, 2001, "Schedule and Scope of Conversion to NUREG-1431, Standard Technical Specifications - Westinghouse Plants," from M. W. Rencheck, Indiana Michigan Power Company, to U. S. Nuclear Regulatory Commission.

Table 3-1: Summary of Technical Specification Changes and NRC Staff Review				
Functional Unit Unit 1 Page # Unit 2 Page #	**Item**	**Description**	**Bounded by NRC-approved Topical Report**	**NRC Staff Comments and Conclusions**
	Bypass Times			
RTS 21 U1 3/4 3-5 U2 3/4 3-4	Reactor Trip Breakers	Extends bypass time from 2+ to 4 hours by removing the reference to note (1) and adding reference to new note (15)	yes[1,2]	Acceptable
	Completion Times			
RTS 21 U1 3/4 3-5 U2 3/4 3-4	Reactor Trip Breakers	Establishes a completion time of 24 hours by removing the reference to note (1) and adding reference to new note (15)	yes[1,2]	Acceptable
	Surveillance Test Intervals (STI) Extended to 62 Days			
RTS 21.A U1 3/4 3-13 U2 3/4 3-12	Reactor Trip Breaker - Shunt Trip Function	Extends STI from 31 to 62 days and modifies applicable Note (5) to test each train at least every other 62 days	yes[1,2]	Acceptable WCAP-15376 and TSTF-411 justify extension to 62 days on a STAGGERED TEST BASIS. As modified Note (5) establishes an equivalent frequency.
RTS 21.B U1 3/4 3-13 U2 3/4 3-12	Reactor Trip Breaker - Undervoltage Trip Function	Extends STI from 31 to 62 days and modifies applicable Note (5) to test each train at least every other 62 days	yes[1,2]	Acceptable WCAP-15376 and TSTF-411 justify extension to 62 days on a STAGGERED TEST BASIS. As modified Note (5) establishes an equivalent frequency.

Table 3-1: Summary of Technical Specification Changes and NRC Staff Review				
Functional Unit Unit 1 Page # Unit 2 Page #	**Item**	**Description**	**Bounded by NRC-approved Topical Report**	**NRC Staff Comments and Conclusions**
RTS 23 U1 3/4 3-13 U2 3/4 3-12	Reactor Trip Bypass Breaker	Extends STI from 31 to 62 days and modifies applicable Note (5) to test each train at least every other 62 days	yes[1,2]	Acceptable WCAP-15376 and TSTF-411 justify extension to 62 days on a STAGGERED TEST BASIS. As modified Note (5) establishes an equivalent frequency.
	Surveillance Test Intervals Extended to 92 Days			
RTS 2 U1 3/4 3-12 U2 3/4 3-11	Power Range, Neutron Flux	Extends STI from 31 to 92 days	yes[1,2]	Acceptable WCAP-15376 and TSTF-411 justify extension to 184 days
RTS 3 U1 3/4 3-12 U2 3/4 3-11	Power Range, Neutron Flux, High Positive Rate	Extends STI from 31 to 92 days	yes[1,2]	Acceptable WCAP-15376 and TSTF-411 justify extension to 184 days
RTS 4 U1 3/4 3-12 U2 3/4 3-11	Power Range, Neutron Flux, High Negative Rate	Extends STI from 31 to 92 days	yes[1,2]	Acceptable WCAP-15376 and TSTF-411 justify extension to 184 days
RTS 19 U1 3/4 3-13 U2 3/4 3-12	Safety Injection Input from ESF	Extends STI from 31 to 92 days and adds applicability of new Note (15) to test each train at least every other 92 days	yes[1,3]	Acceptable

Table 3-1: Summary of Technical Specification Changes and NRC Staff Review				
Functional Unit Unit 1 Page # Unit 2 Page #	**Item**	**Description**	**Bounded by NRC-approved Topical Report**	**NRC Staff Comments and Conclusions**
RTS 22 U1 3/4 3-13 U2 3/4 3-12	Automatic Trip Logic - Reactor Protection System	Extends STI from 31 to 92 days and adds applicability of new Note (15) to test each train at least every other 92 days	yes[1,2]	Acceptable WCAP-15376 and TSTF-411 justify extension to 92 days on a STAGGERED TEST BASIS. As modified Note (15) establishes a frequency equivalent to 92 days on a STAGGERED TEST BASIS.
ESF 1.b U1 3/4 3-31 U2 3/4 3-30	Safety Injection, Turbine Trip, Feedwater Isolation, and Motor Driven Auxiliary Feedwater Pumps - Automatic Actuation Logic	Extends STI from 31 to 92 days and modifies applicable Note (2) to test each train at least every other 92 days	yes[1,2]	Acceptable WCAP-15376 and TSTF-411 justify extension to 92 days on a STAGGERED TEST BASIS. As modified Note (2) establishes an equivalent frequency.
ESF 2.b U1 3/4 3-31 U2 3/4 3-30	Containment Spray - Automatic Actuation Logic	Extends STI from 31 to 92 days and modifies applicable Note (2) to test each train at least every other 92 days	yes[1,2]	Acceptable WCAP-15376 and TSTF-411 justify extension to 92 days on a STAGGERED TEST BASIS. As modified Note (2) establishes an equivalent frequency.
ESF 3.a. 2) U1 3/4 3-32 U2 3/4 3-30	Containment Isolation - Phase “A” Isolation - From Safety Injection Automatic Actuation Logic	Extends STI from 31 to 92 days and modifies applicable Note (2) to test each train at least every other 92 days	yes[1,2]	Acceptable WCAP-15376 and TSTF-411 justify extension to 92 days on a STAGGERED TEST BASIS. As modified Note (2) establishes an equivalent frequency.

Table 3-1: Summary of Technical Specification Changes and NRC Staff Review				
Functional Unit Unit 1 Page # Unit 2 Page #	**Item**	**Description**	**Bounded by NRC-approved Topical Report**	**NRC Staff Comments and Conclusions**
ESF 3.b. 2) U1 3/4 3-32 U2 3/4 3-30	Containment Isolation - Phase “B” Isolation - Automatic Actuation Logic	Extends STI from 31 to 92 days and modifies applicable Note (2) to test each train at least every other 92 days	yes[1,2]	Acceptable WCAP-15376 and TSTF-411 justify extension to 92 days on a STAGGERED TEST BASIS. As modified Note (2) establishes an equivalent frequency.
ESF 4.b U1 3/4 3-33 U2 3/4 3-31	Steam Line Isolation - Automatic Actuation Logic	Extends STI from 31 to 92 days and modifies applicable Note (2) to test each train at least every other 92 days	yes[1,2]	Acceptable WCAP-15376 and TSTF-411 justify extension to 92 days on a STAGGERED TEST BASIS. As modified Note (2) establishes an equivalent frequency.
ESF 6.c U1 3/4 3-33 U2 3/4 3-31	Motor Driven Auxiliary Feedwater Pumps - Safety Injection	Extends STI from 31 to 92 days and modifies applicable Note (2) to test each train at least every other 92 days	yes[1,2]	Acceptable WCAP-15376 and TSTF-411 justify extension to 184 days
ESF 10.b.c U1 3/4 3-33b U2 3/4 3-32	Containment Air Recirculation Fan - Automatic Actuation Logic	Extends STI from 31 to 92 days and modifies applicable Note (2) to test each train at least every other 92 days	yes[1]	Acceptable[3] WCAP-15376 justifies extension to 184 days
	Surveillance Test Intervals Extended to 184 Days			
RTS 5 U1 3/4 3-12 U2 3/4 3-11	Intermediate Range, Neutron Flux	Extendeds interval for re-performance from 7 to 184 days adding new Note (17) to reflect this requirement	yes[1,2]	Acceptable
RTS 7 U1 3/4 3-12 U2 3/4 3-11	Overtemperature delta-T	Extends STI from 31 to 184 days	yes[1,2]	Acceptable

Table 3-1: Summary of Technical Specification Changes and NRC Staff Review				
Functional Unit Unit 1 Page # Unit 2 Page #	**Item**	**Description**	**Bounded by NRC-approved Topical Report**	**NRC Staff Comments and Conclusions**
RTS 8 U1 3/4 3-12 U2 3/4 3-11	Overpower delta-T	Extends STI from 31 to 184 days	yes[1,2]	Acceptable
RTS 9 U1 3/4 3-12 U2 3/4 3-11	Pressurizer Pressure - Low	Extends STI from 31 to 184 days	yes[1,2]	Acceptable
RTS 10 U1 3/4 3-12 U2 3/4 3-11	Pressuurizer Pressure - High	Extends STI from 31 to 184 days	yes[1,2]	Acceptable
RTS 11 U1 3/4 3-12 U2 3/4 3-11	Pressurizer Water Level - High	Extends STI from 31 to 184 days	yes[1,2]	Acceptable
RTS 12 U1 3/4 3-12 U2 3/4 3-11	Loss of Flow-Single Loop	Extends STI from 31 to 184 days	yes[1,2]	Acceptable
RTS 14 U1 3/4 3-13 U2 3/4 3-12	Steam Generator Water Level - Low-Low	Extends STI from 31 to 184 days	yes[1,2]	Acceptable
RTS 15 U1 3/4 3-13 U2 3/4 3-12	Steam / Feedwater Flow Mismatch and Low Steam Generator Water Level	Extends STI from 31 to 184 days	yes[1,2]	Acceptable
ESF 1.c U1 3/4 3-31 U2 3/4 3-30	Safety Injection, Turbine Trip, Feedwater Isolation, and Motor Driven Auxiliary Feedwater Pumps - Containment Pressure - High	Extends STI from 31 to 184 days	yes[1,2]	Acceptable

Table 3-1: Summary of Technical Specification Changes and NRC Staff Review				
Functional Unit Unit 1 Page # Unit 2 Page #	**Item**	**Description**	**Bounded by NRC-approved Topical Report**	**NRC Staff Comments and Conclusions**
ESF 1.d U1 3/4 3-31 U2 3/4 3-30	Safety Injection, Turbine Trip, Feedwater Isolation, and Motor Driven Auxiliary Feedwater Pumps - Pressurizer Pressure - Low	Extends STI from 31 to 184 days	yes[1,2]	Acceptable
ESF 1.e U1 3/4 3-31 U2 3/4 3-30	Safety Injection, Turbine Trip, Feedwater Isolation, and Motor Driven Auxiliary Feedwater Pumps - Differential Pressure Between Steam Lines - High	Extends STI from 31 to 184 days	yes[1,2]	Acceptable
ESF 1.f U1 3/4 3-31 U2 3/4 3-30	Acceptable Safety Injection, Turbine Trip, Feedwater Isolation, and Motor Driven Auxiliary Feedwater Pumps -Steam Line Pressure - Low	Extends STI from 31 to 184 days	yes[1,2]	Acceptable
ESF 2.c U1 3/4 3-31 U2 3/4 3-30	Containment Spray - Containment Pressure - High - High	Extends STI from 31 to 184 days	yes[1,2]	Acceptable

Table 3-1: Summary of Technical Specification Changes and NRC Staff Review				
Functional Unit Unit 1 Page # Unit 2 Page #	**Item**	**Description**	**Bounded by NRC-approved Topical Report**	**NRC Staff Comments and Conclusions**
ESF 3.b.3) U1 3/4 3-32 U2 3/4 3-30	Containment Isolation - Phase "B" Isolation - Containment Pressure - High-High	Extends STI from 31 to 184 days	yes[1,2]	Acceptable
ESF 4.c U1 3/4 3-33 U2 3/4 3-31	Steam Line Isolation - Containment Pressure - High-High	Extends STI from 31 to 184 days	yes[1,2]	Acceptable
ESF 4.d U1 3/4 3-33 U2 3/4 3-31	Steam Line Isolation - Steam Flow in Two Steam Lines Coincident with T_{avg} - Low-Low	Extends STI from 31 to 184 days	yes[1,2]	Acceptable
ESF 4.e U1 3/4 3-33 U2 3/4 3-31	Steam Line Isolation - Steam Line Pressure - Low	Extends STI from 31 to 184 days	yes[1,2]	Acceptable
ESF 5.a U1 3/4 3-33 U2 3/4 3-31	Turbine Trip and Feedwater Isolation - Steam Generator Water Level - High-High	Extends STI from 31 to 184 days	yes[1,2]	Acceptable
ESF 6.a U1 3/4 3-33 U2 3/4 3-31	Motor Driven Auxiliary Feedwater Pumps - Steam Generator Water Level - Low-Low	Extends STI from 31 to 184 days	yes[1,2]	Acceptable
ESF 7.a U1 3/4 3-33a U2 3/4 3-32	Turbine Driven Auxiliary Feedwater Pumps - Steam Generator Water Level - Low-Low	Extends STI from 31 to 184 days	yes[1,2]	Acceptable

Table 3-1: Summary of Technical Specification Changes and NRC Staff Review				
Functional Unit Unit 1 Page # Unit 2 Page #	**Item**	**Description**	**Bounded by NRC-approved Topical Report**	**NRC Staff Comments and Conclusions**
ESF 10.c U1 3/4 3-33b U2 3/4 3-32	Containment Air Recirculation Fan - Containment Pressure - High	Extends STI from 31 to 184 days	yes[1],[4]	Acceptable
Administrative or Editorial				
Definitions U1 1-9 U2 1-10	Definition	Addition of a “2 months” frequency Notation consisting of 62 days	no	Acceptable Administrative addition of definition to support extended surveillance intervals accepted by WCAP-15376 and TSTF-411, within the framework of the existing technical specifications.
RTS 21 U1 3/4 3-5 U2 3/4 3-4	Reactor Trip Bypass Breaker	Increase the spacing between the lines for MODES 1, 2 and MODES 3*, 4*, 5* Delete the comma after “2" in the APPLICABLE MODES column (Unit 2 only)	no	Acceptable These editorial changes more clearly indicate that ACTIONs 13 and 15 apply only in MODES 1 and 2 and that ACTION 14 only applies in MODES 3, 4, and 5. There is no change in intended action as a result of this change.
RTS 22 U1 3/4 3-5 U2 3/4 3-4	Automatic Trip Logic - Reactor Protection System	Increase the spacing between the lines for MODES 1, 2 and MODES 3*, 4*, 5*	no	Acceptable This editorial change more clearly indicates that ACTION 1 applies only in MODES 1 and 2 and that ACTION 14 only applies in MODES 3, 4, and 5. There is no change in intended action as a result of this change.

Table 3-1: Summary of Technical Specification Changes and NRC Staff Review				
Functional Unit Unit 1 Page # Unit 2 Page #	**Item**	**Description**	**Bounded by NRC-approved Topical Report**	**NRC Staff Comments and Conclusions**
RTS Notation U1 3/4 3-14 U2 3/4 3-13	Notation (16)	Addition of Note (16) with an annotation of “not used”	no	Acceptable This note was proposed in the original application for amendment dated August 30, 2002. It was modified by supplement dated February 27, 2003. The reference to the note is being retained as a placekeeper.

Table 3-1 Notes:

1. WCAP-15376-P, Rev. 0, “Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times.“

2. Industry/TSTF Standard Technical Specification Change Traveler, TSTF-411, Rev. 1, “Surveillance Test Interval Extentions for Components of the Reactor Protection System (WCAP-15376-P),” dated August 7, 2002.

3. Although this functional unit is not explicitly evaluated in WCAP-15376-P. The testing required for this functional unit is satisfied during the performance of required testing specified for ESFAS functional units 1.c, 1.d, and 1.e. WCAP -15376-P and TSTF-411, Rev. 1 accept the extension, from 31 to 184 days, of the testing frequency for these ESFAS functional units. Thus the extension of the frequency for testing functional unit 19 is adequately justified and acceptable.

4. Although this functional unit is not explicitly evaluated in WCAP-15376-P. It uses a master relay and slave relays of the same type as others used in the ESF system. This equipment is consistent with representative equipment and actuation logic specifically evaluated in WCAP-15376-P.

3.1 SUMMARY

In summary, the NRC staff, having reviewed the licensee’s assessment of the proposed TS changes, concludes that the changes are bounded by the NRC staff safety evaluation documenting acceptance of WCAP-15376-P Rev. 0 and TSTF-411 Rev. 1 or are justified administrative or editorial changes. Therefore, the NRC staff finds the proposed changes to be acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

These amendments change the requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or change the surveillance requirements. The staff has 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 the amendments involve no significant hazards consideration and there has been no public comment on such finding (67 FR 63695). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

Principal Contributor:

H. K. Chernoff

Date: May 23, 2003