UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 11, 2015

Mr. Lawrence J. Weber
Senior Vice President and
Chief Nuclear Officer
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

SUBJECT: DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: TECHNICAL SPECIFICATIONS SURVEILLANCE REQUIREMENTS 3.8.1.10, 3.8.1.11, AND 3.8.1.15 (CAC NOS. MF5436 AND MF5437)

Dear Mr. Weber:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 330 to Renewed Facility Operating License No. DPR-58, and Amendment No. 311 to Renewed Facility Operating License No. DPR-74, for the Donald C. Cook Nuclear Plant (CNP), Units 1 and 2, respectively. The amendments consist of changes to the technical specifications (TSs) in response to your application dated December 17, 2014, as supplemented by letters dated July 9, 2015 and October 30, 2015.

The amendments revise the CNP Units 1 and 2 TS 3.8.1, "AC Sources – Operating," to allow surveillance testing of the onsite standby emergency diesel generators (EDGs) during modes in which it was previously restricted. Specifically, the changes remove the mode restrictions in the notes of the surveillance requirements 3.8.1.10, EDG single largest load rejection test, 3.8.1.11, EDG full load rejection test, and 3.8.1.15, EDG endurance run.

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

Sincerely,

Allison W. Dietrich, Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures:
1. Amendment No. 330 to DPR-58
2. Amendment No. 311 to DPR-74
3. Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-315

DONALD C. COOK NUCLEAR PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 330
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) dated December 17, 2014, as supplemented by letters dated July 9, 2015 and October 30, 2015, 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 Renewed Facility Operating License No. DPR-58 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, as revised through Amendment

No. 330, are hereby incorporated in this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 140 days.

FOR THE NUCLEAR REGULATORY COMMISSION

David L. Pelton, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to Renewed Facility Operating License No. DPR-58 and Technical Specifications

Date of Issuance: December 11, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 330

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-58

DOCKET NO. 50-315

Replace the following page of the Renewed Facility Operating License No. DPR-58 with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

REMOVE	INSERT
- 3 -	- 3 -

Replace the following pages of 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
3.8.1-9	3.8.1-9
3.8.1-10	3.8.1-10
3.8.1-13	3.8.1-13

and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument and equipment calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not to exceed 3304 megawatts thermal in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 330, are hereby incorporated in this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Less than Four Loop Operation

The licensee shall not operate the reactor at power levels above P-7 (as defined in Table 3.3.1-1 of Specification 3.3.1 of Appendix A to this renewed operating license) with less than four reactor coolant loops in operation until (a) safety analyses for less than four loop operation have been submitted, and (b) approval for less than four loop operation at power levels above P-7 has been granted by the Commission by amendment of this license.

(4) Fire Protection Program

Indiana Michigan Power Company shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee's amendment request dated July 1, 2011, as supplemented by letters dated September 2, 2011, April 27, 2012, June 29, 2012, August 9, 2012, October 15, 2012, November 9, 2012, January 14, 2013, February 1, 2013,

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.10	------------------NOTES------------------ 1. If performed with the DG synchronized with offsite power, it shall be performed at a power factor ≤ 0.86. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. --- Verify each DG rejects a load greater than or equal to its associated single largest post-accident load, and: a. Following load rejection, the frequency is ≤ 64.4 Hz; b. Within 2 seconds following load rejection, the voltage is ≥ 3910 V and ≤ 4400 V; and c. Within 2 seconds following load rejection, the frequency is ≥ 59.4 Hz and ≤ 60.5 Hz.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.11	-----NOTES----- 1. If performed with DG synchronized with offsite power, it shall be performed at a power factor ≤ 0.86. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. ---------- Verify each DG does not trip and voltage is maintained ≤ 5350 V during and following a load rejection of ≥ 3150 kW and ≤ 3500 kW.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.14	-----------------------------NOTE------------------------------ This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the unit is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. --- Verify each DG's automatic trips are bypassed on an actual or simulated loss of voltage signal on the emergency bus or an actual or simulated ESF actuation signal except: a. Engine overspeed; and b. Generator differential current.	24 months
SR 3.8.1.15	-----------------------------NOTES---------------------------- 1. Momentary transients outside the load and power factor ranges do not invalidate this test. 2. If performed with DG synchronized with offsite power, it shall be performed at a power factor ≤ 0.86. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. --- Verify each DG operates for ≥ 8 hours at a load ≥ 3150 kW and ≤ 3500 kW.	24 months

UNITED STATES
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INDIANA MICHIGAN POWER COMPANY

DOCKET NO. 50-316

DONALD C. COOK NUCLEAR PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 311
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) dated December 17, 2014, as supplemented by letters dated July 9, 2015 and October 30, 2015, 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 Renewed Facility Operating License No. DPR-74 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, as revised through Amendment

No. 311, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 140 days.

FOR THE NUCLEAR REGULATORY COMMISSION

David L. Pelton, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to Renewed Facility Operating License No. DPR-74 and Technical Specifications

Date of Issuance: December 11, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 311

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-74

DOCKET NO. 50-316

Replace the following page of the Renewed Facility Operating License No. DPR-74 with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

REMOVE	INSERT
- 3 -	- 3 -

Replace the following pages of 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
3.8.1-9	3.8.1-9
3.8.1-10	3.8.1-10
3.8.1-13	3.8.1-13

radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument and equipment calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not to exceed 3468 megawatts thermal in accordance with the conditions specified herein and in Attachment 1 to the renewed operating license. The preoperational tests, startup tests and other items identified in Attachment 1 to this renewed operating license shall be completed. Attachment 1 is an integral part of this renewed operating license.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 311, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Additional Conditions

(a) Deleted by Amendment No. 76

(b) Deleted by Amendment No. 2

(c) Leak Testing of Emergency Core Cooling System Valves

Indiana Michigan Power Company shall prior to completion of the first inservice testing interval leak test each of the two valves in series in the

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.10	-----------------------------NOTES----------------------------- 1. If performed with the DG synchronized with offsite power, it shall be performed at a power factor ≤ 0.86. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. --- Verify each DG rejects a load greater than or equal to its associated single largest post-accident load, and: a. Following load rejection, the frequency is ≤ 64.4 Hz; b. Within 2 seconds following load rejection, the voltage is ≥ 3910 V and ≤ 4400 V; and c. Within 2 seconds following load rejection, the frequency is ≥ 59.4 Hz and ≤ 60.5 Hz.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.11	-------------------------------NOTES----------------------------- 1. If performed with DG synchronized with offsite power, it shall be performed at a power factor ≤ 0.86. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. --- Verify each DG does not trip and voltage is maintained ≤ 5350 V during and following a load rejection of ≥ 3150 kW and ≤ 3500 kW.	24 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.14	-----------------------------NOTE------------------------------ This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the unit is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR. --- Verify each DG's automatic trips are bypassed on an actual or simulated loss of voltage signal on the emergency bus or an actual or simulated ESF actuation signal except: a. Engine overspeed; and b. Generator differential current.	24 months
SR 3.8.1.15	-----------------------------NOTES---------------------------- 1. Momentary transients outside the load and power factor ranges do not invalidate this test. 2. If performed with DG synchronized with offsite power, it shall be performed at a power factor ≤ 0.86. However, if grid conditions do not permit, the power factor limit is not required to be met. Under this condition the power factor shall be maintained as close to the limit as practicable. --- Verify each DG operates for ≥ 8 hours at a load ≥ 3150 kW and ≤ 3500 kW.	24 months

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 330 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-58

AND

AMENDMENT NO. 311 TO RENEWED 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 December 17, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML14356A022), as supplemented by letters dated July 9, 2015 and October 30, 2015 (ADAMS Accession Nos. ML15195A434 and ML15308A094, respectively), Indiana Michigan Power Company (I&M, the licensee) requested a license amendment for the Donald C. Cook Nuclear Plant (CNP), Units 1 and 2. Specifically, the licensee requested to revise the CNP Units 1 and 2 technical specification (TS) 3.8.1, "AC Sources – Operating," to allow surveillance testing of the onsite standby emergency diesel generators (EDGs) during modes in which it was previously restricted. The proposed changes would remove the mode restrictions in the notes of the surveillance requirements (SRs) 3.8.1.10, EDG single largest load rejection test, 3.8.1.11, EDG full load rejection test, and 3.8.1.15, EDG endurance run.

The supplemental letters dated July 9, 2015, and October 30, 2015, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on March 17, 2015 (80 FR 13909).

2.0 REGULATORY EVALUATION

The construction permits for CNP were issued, and the majority of construction was completed prior to issuance of Appendix A, General Design Criteria (GDC), to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, in 1971 by the Atomic Energy Commission. However, CNP was designed and constructed to comply with the proposed GDC criteria published on

July 11, 1967 (32 FR 10213). Section 1.4 of the CNP Updated Final Safety Analysis Report (UFSAR), "Plant Specific Design Criteria" (PSDC), defines the principal criteria and safety objectives for the design of CNP. The current Appendix A of 10 CFR Part 50 GDC differ both in numbering and content from the PSDC for CNP.

The NRC's acceptance criteria for this license amendment request (LAR) are based on the following regulations and guidance documents:

- 10 CFR Part 50.36, "Technical Specifications," requires, in part, that the operating license of a nuclear production facility include TSs. Section 50.36(c)(3) of 10 CFR requires that the TSs include SRs, which are requirements "relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, that the limiting conditions for operation will be met."

- CNP PSDC 39, "Emergency Power" states, "An emergency power source shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning of the engineered safety features and protection systems required to avoid undue risk to the health and safety of the public. This power source shall provide this capacity assuming a failure of a single active component."

- Regulatory Guide (RG) 1.9, Revision 3, "Selection, Design, Qualification, and Testing of Emergency Diesel Generator Units Used as Class 1E Onsite Electric Power Systems at Nuclear Power Plants" (ADAMS Accession No. ML003739929), provides guidance for EDG design and testing. This guidance is used to develop the SRs for EDGs.

- NUREG-1431, Revision 2, "Standard Technical Specification Westinghouse Plants" (ADAMS Accession No. ML12100A222) contains the improved Standard Technical Specifications (STS) for Westinghouse plants. The improved STS were developed based on the criteria in the Final Commission Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, dated July 22, 1993, which was subsequently codified by changes to 10 CFR 50.36 (60 FR 36953).

3.0 TECHNICAL EVALUATION

3.1 Description of Emergency Power System

As stated in Chapter 8 of the CNP UFSAR, the auxiliary power of each CNP unit is distributed from the 4.16 kilovolts (KV) switchgear, which is energized from the main generator through unit auxiliary transformers (UATs) during normal operation, and from the preferred offsite power source reserve auxiliary transformers (RATs) during start-up or shutdown operations. During shutdown and outage conditions, the 4.16 KV buses are also capable of being energized from the offsite power source, through the main power transformer and the UATs when the generator is off-line and the generator disconnect links are removed.

The emergency power source for each unit consists of two 4.16 KV, 3,500 kilowatt (KW) EDGs. Each EDG supplies power to two safety-related 4.16 KV buses, and is sized to power one train of safety equipment, assuming a loss-of-offsite-power (LOOP) concurrent with a loss-of-coolant accident (LOCA), with or without containment spray (CS).

The CNP auxiliary safety and non-safety load buses are connected radially to either the RATs or UATs. Each RAT feeds two 4.16 KV balance of plant (BOP) buses that feed large BOP motor loads such as the reactor coolant pumps and circulating water pumps. The 4.16 KV BOP buses then feed the 4.16 KV safety buses, which are connected to the EDG via EDG output breakers.

Each 4.16 KV safety bus is equipped with a set of three undervoltage, or loss of voltage (LOV), relays configured in a 2/3 logic. The LOV relays generate a LOOP signal when the voltage on a safety bus is less than approximately 78 percent of nominal bus voltage for a duration of 2 seconds. The LOOP signal actuates a series of auxiliary relays (master load shedding relays) that initiates load shedding and provide a permissive for the closing of the EDG output breakers. The 2 second time delay on the LOV relays allows for motor voltage decay of running motors and overrides any momentary grid transient before the EDG is connected to the bus.

Each of the 4.16 KV safety buses – T11A and T11D for Unit 1, T21A and T21D for Unit 2 – is equipped with a set of three degraded voltage relays (DVRs) configured in a 2/3 logic to protect large motors from a sustained low or degraded voltage condition. The DVRs actuate when the voltage on these buses falls below approximately 94 percent of nominal bus voltage for a duration of 2 minutes (with no concurrent accident signal) or 9 seconds (with a concurrent Safety Injection or Steam Generator Lo-Lo signal).

Upon sensing a loss of voltage, master (auxiliary) relays operated by the LOV relays automatically start the EDGs, and trip the normal feed circuit breakers for the 4.16 KV buses. The master relays also trip all motor feeder breakers and 480 volt (V) bus transformer feeder breakers on the buses, the 600 V bus tie breaker, non-essential 600 V feeder breakers, and 480 V bus breakers. Each EDG comes up to speed and is capable of accepting load within 10 seconds. The EDG bus input circuit breakers, which connect the EDG output to the 4.16 KV bus system, are automatically closed when the EDG voltage and speed approach rated values. The EDGs supply power to the safety-related 4.16 KV buses and their associated 600 V buses and transformers. A Safety Injection signal also starts the EDGs. To avoid overloading the EDGs, all loads are shed when the Safety Injection signal occurs, and the safety-related 4.16 KV buses are energized from the EDGs. The safety loads are subsequently loaded in a sequential manner, as required.

The emergency power system and the EDGs are equipped with monitors and annunciators to ensure that the operators have adequate information on system status. Suitable protective devices are provided to initiate prompt automatic detection and isolation of defective or faulted equipment. All annunciators and protective devices are in service as applicable during EDG testing. Only the EDG differential protection and overspeed trips are operative during actual or simulated emergency conditions. The EDGs are periodically tested to ensure availability of standby power. During testing of a given EDG system, the redundant EDG train system remains available in case it is needed for emergency conditions.

The TS SRs 3.8.1.10, 3.8.1.11, and 3.8.1.15 demonstrate the operability of the EDGs. These SRs are currently normally prohibited to be performed in Mode 1 or 2. Mode 1 is power operation with rated thermal power (RTP) greater than 5 percent. Mode 2 is a startup condition with RTP less than or equal to 5 percent.

3.2 Proposed Technical Specification Changes

The licensee proposed to delete the Mode 1 and 2 surveillance testing restrictions for SRs 3.8.1.10, 3.8.1.11, and 3.8.1.15.

SR 3.8.1.10 verifies that each EDG rejects a load greater than or equal to its associated single largest post-accident load and verifies that:

a. following load rejection, the frequency is ≤ 64.4 Hz;
b. within 2 seconds following load rejection, the voltage is ≥ 3910 V and ≤ 4400 V; and
c. within 2 seconds following load rejection, the frequency is ≥ 59.4 Hz and ≤ 60.5 Hz.

SR 3.8.1.11 verifies that each EDG does not trip and that voltage is maintained ≤ 5350 V during and following a load rejection of ≥ 3150 KW and ≤ 3500 KW.

SR 3.8.1.15 verifies that each EDG operates for ≥ 8 hours at a load ≥ 3150 KW and ≤ 3500 KW.

The above SRs currently include a Note (Note 1 for SRs 3.8.1.10 and 3.8.1.11; Note 2 for SR 3.8.1.15) that states: "This Surveillance shall not normally be performed in MODE 1 or 2. However, this Surveillance may be performed to reestablish OPERABILITY provided an assessment determines the safety of the unit is maintained or enhanced. Credit may be taken for unplanned events that satisfy this SR."

The proposed changes would delete the above Note from the SRs to allow for SR performance in Modes 1 and 2.

3.3 Normal Plant Operation

As stated in its October 30, 2015 supplement, the licensee defines a non-emergency condition as a normal mode of operation with no LOOP or LOCA signal standing, while an emergency condition is a LOCA or LOOP event. This section of the safety evaluation discusses the NRC staff's evaluation of the plant response during the performance of SRs 3.8.1.10, 3.8.1.11, and 3.8.1.15, in Modes 1 and 2, during non-emergency conditions.

As stated in the December 17, 2014 LAR, SR 3.8.1.10, the EDG single largest load rejection test, is currently performed while the reactor is in Mode 5 or 6. Mode 5 is a cold shutdown condition with reactor coolant temperature less than 200 degrees Fahrenheit. Mode 6 is a refueling condition, with one or more reactor vessel head closure bolts less than fully tensioned. The test is performed by paralleling the EDG with offsite power, manually raising the EDG to the required load, and then opening the EDG output breaker. Opening the EDG output breaker simulates the single largest load rejection. The EDG is separated from its associated safety-related 4.16 KV bus, and the offsite circuit continues to supply that bus.

The EDG full load rejection test, SR 3.8.1.11, is currently performed by paralleling the EDG with offsite power while the reactor is in Mode 5 or 6, manually raising the EDG to the required 100 percent load, and then opening the EDG output breaker. Opening the EDG output breaker simulates the full load rejection. The EDG is separated from its associated safety-related 4.16 KV bus, and the offsite circuit continues to supply that bus.

In SR 3.8.1.15, the EDG endurance test run, is currently performed by paralleling the EDG with offsite power while the reactor is in Mode 5 or 6, and manually raising the EDG loading to the required loads (90-100 percent of the EDG continuous load rating). In its supplement dated October 30, 2015, the licensee stated that during this eight hour run, the EDG is connected only to the safety-related 4.16 KV bus that has the large motor loads (T11A or T11D for Unit 1, T21A or T21D for Unit 2) when paralleled with the offsite power source. The other bus (T11B or T11C for Unit 1, T21B or T21C for Unit 2), is briefly paralleled with the EDG at the end of the test. During an emergency, both output breakers to the buses would close and the EDG would be connected to both safety-related 4.16 KV buses.

One concern associated with performing SRs 3.8.1.10, 3.8.1.11, and 3.8.1.15 in Mode 1 or 2 is that performance of the SRs could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. The restriction from normally performing the surveillances in Mode 1 or 2 makes an allowance for the surveillances to be performed for the purpose of reestablishing operability (e.g., post work testing following corrective maintenance, corrective modification, deficient or incomplete surveillance testing, and other unanticipated operability concerns), provided an assessment determines plant safety is maintained or enhanced. This assessment shall, at a minimum, consider the potential outcomes and transients associated with a failed surveillance, a successful surveillance, and a perturbation of the offsite or onsite system when they are tied together or operated independently for the surveillance, as well as the operator procedures available to cope with these outcomes. These shall be measured against the avoided risk of a plant shutdown and startup to determine that plant safety is maintained or enhanced when the surveillance is performed in Mode 1 or 2.

The licensee provided the following discussion in the LAR to justify their proposed removal of mode restrictions for the above SRs.

- Any grid disturbances would only possibly affect one EDG because only one EDG per unit is paralleled to offsite power or generator output at a time. Protective relaying protects the EDG being tested from equipment damage. In the event of a grid disturbance, the EDG output breaker trips to separate the EDG from its associated safety-related 4.16 KV bus; the EDG could be restored via operator action. If the disturbance affects the EDG being tested, the redundant EDG will remain operable to power the redundant train of onsite alternating current (AC) power, which is fully capable of mitigating a design basis accident or providing for safe shutdown of the associated unit. To minimize the likelihood of a power disturbance or fault related to switchyard activities, the following statements will be added to the TS Bases to prohibit switchyard work activities during the performance of SR 3.8.1.10, SR 3.8.1.11, and SR 3.8.1.15 in Mode 1 or 2:

  > Prior to the performance of this Surveillance in MODE 1 or 2 a risk assessment shall be performed to determine that plant safety is maintained. As part of this assessment weather conditions will be

assessed, and the SR will not be scheduled when severe weather conditions and/or unstable grid conditions are predicted or present. Also, no discretionary maintenance activities will be scheduled which could cause a line outage or challenge offsite power availability. Additionally, no switchyard activities will be allowed during the performance of this surveillance.

- CNP online risk management considerations ensure that these SRs are not scheduled during periods where there is increased potential for grid or bus disturbance (storms, grid emergencies, etc.).

- There are no recorded voltage oscillations when SR 3.8.1.10 is performed at shutdown at CNP. The voltage change is a smooth step change, and the voltage recovery is within 2 seconds. Thus, the performance of SR 3.8.1.10 in any modes would not cause a significant electrical perturbation that would adversely affect the onsite AC electrical system.

- While the EDG is in test mode, non-emergency trip features are in effect to protect it from equipment damage. If an emergency demand, such as engineered safety feature or LOOP signal, occurs during the test, the EDG will automatically revert to the emergency mode and bypass these trips. Thus, these additional trip features are not a significant concern during the performance of these SRs. The operators receive indication and alarms in the control room that the preferred power source is lost.

- CNP currently tests the EDGs paralleled to offsite power or generator output while at power during required monthly surveillance testing, SR 3.8.1.3, the EDG monthly surveillance test run, with no adverse effect. The proposed test configuration for SR 3.8.1.10, SR 3.8.1.11, and SR 3.8.1.15 in Mode 1 will be similar to that of SR 3.8.1.3. SR 3.8.1.3 verifies that the EDG is capable of synchronizing with the offsite power electrical system and accepting loads of 90 percent to 100 percent of the continuous rating of the EDG for a run time of at least 60 minutes.

- When an EDG is undergoing any of these SRs in Mode 5 or 6, it is considered inoperable but available. The EDG will remain available during the tests in Mode 1 or 2. As a result, there is no increase in unavailability of the EDG, and there is minimal increase in the risk.

- Historical review of EDG starts at CNP, as well as a search for the CNP corrective action program for the last 10 years, has shown no instance of additional problems caused by the starts of the EDGs to the EDGs themselves, other equipment, or the electrical grid, regardless of the mode in which the EDG was actually started.

In its supplement dated July 9, 2015, the licensee provided copies of CNP procedures that include the recorded data monitored during the last performance of the EDG load rejection tests, SR 3.8.1.10 and SR 3.8.1.11, for both units. The NRC staff finds that the recorded load, voltage, and frequency data are well within the required ranges in the SRs.

Also in its supplement dated July 9, 2015, the licensee provided the summary of the results for the EDG full load rejection simulations performed using ETAP software and the CNP control room simulator. Based on ETAP analysis, the licensee stated that the load flow analysis shows that the bus voltage reduction due to the full load rejection is approximately 3 percent when the 4.16 KV buses are fed from the UAT, and 1.7 percent when the 4.16 KV buses are fed from the RATs. Based on the control room simulation, the licensee stated that the voltage transient on the 4.16 KV buses was negligible due to compensation from the main generator voltage regulator response when the 4.16 KV buses are fed from the UAT. The voltage transient was approximately 1 percent when the 4.16 KV buses are fed from the RATs. The licensee further stated that the buses are normally operated at 100 percent rated voltage, and the connected motors are rated to operate down to 90 percent voltage level. Based on this information, the NRC staff finds that the expected voltage reduction on the 4.16 KV bus will not adversely impact the operation of connected motors loads.

3.3.1 Protective Devices for Emergency Diesel Generators

In its supplement dated July 9, 2015, the licensee provided a discussion of all electrical protective devices provided for the EDGs. The licensee stated that the EDG electrical protective devices, namely generator overcurrent, generator differential, and generator neutral overcurrent relays, are active during non-emergency conditions. The licensee also stated that the trip signals for electronic overspeed, generator differential, and emergency trip pushbutton are effective during an emergency automatic fast start. In its supplement dated October 30, 2015, the licensee clarified that the overcurrent relays remain active during test mode, but are bypassed during emergency conditions, and that the trip function for differential relays remains active during test mode and emergency conditions. Based on this information, the staff finds that the EDG will have adequate protection during performance of the above SRs.

In its supplement dated July 9, 2015, the licensee provided a summary of the impact of an increase in short circuit current on the switchgear ratings and on the protection settings due to the testing in Mode 1 or 2. The licensee stated that the maximum fault current of 42 kilo amperes (KA) calculated at the safety-related 4.16 KV bus with the EDG in parallel does not exceed the fault rating (46.9 KA) of the 4.16 KV switchgear and circuit breakers. The fault currents are conservatively estimated based on the low circuit impedances and maximum fault contribution from the system and the motors connected to the bus. The EDG overcurrent relays are set to pick up at 960 amperes in 1.5 seconds and have fault thermal rating of over 60 KA for 1 second. The EDG overload relays are set to coordinate with the instantaneous phase overcurrent relays on the 4.16 KV motors to prevent false tripping. The NRC staff reviewed the above information and finds that the switchgear, breakers, and EDG overcurrent relays are adequately rated to protect the safety-related equipment during a fault while the EDG is being tested in Mode 1 or 2.

3.3.2 Upstream Fault

In its supplement dated July 9, 2015, the licensee provided an analysis for a fault on any bus upstream of the 4.16 KV bus connected to the EDG, while the EDG is paralleled to the offsite source, with the EDG bus fed from the UAT or the RAT in Mode 1 or 2.

For a fault at the upstream 4.16 KV reactor coolant pump bus, there may be a relay race between the overcurrent relays protecting the EDG, and overcurrent relays protecting the main source, which would be either UAT or the RAT. The overcurrent relays will isolate the fault by tripping the main feed breakers to the reactor coolant pump bus and EDG output breaker. Regardless of which source's overcurrent relay actuates first, the fault will remain energized until both sources are disconnected by their overcurrent protection. The overcurrent relay on the EDG will also lockout the EDG and require manual reset as stated above. Once the 4.16 KV buses are deenergized, the LOV relays will initiate the normal load shed and EDG restart. The faulted reactor coolant pump bus will be isolated from the safety buses by the load shed circuit response. Following a manual reset of the lockout relay, the safety bus will be automatically reloaded.

Based on its review of the above explanation, the NRC staff finds that a fault on the upstream 4.16 KV bus during the EDG testing will be adequately isolated, and the EDG will be adequately protected.

3.3.3 Normal Plant Operation Conclusion

The NRC staff reviewed the information provided by the licensee regarding data from previous SRs, load rejection simulations, EDG protective devices, and upstream faults. Based on its review of this information, the staff finds that performance of the EDG SRs in Modes 1 and 2 will not have an adverse impact on the onsite AC electrical systems during normal plant operation.

3.4 Emergency Conditions

This section of the safety evaluation discusses the NRC staff's evaluation of the plant response during the performance of SRs 3.8.1.10, 3.8.1.11, and 3.8.1.15, in Modes 1 and 2, if an emergency condition occurs. In its supplements dated July 9, 2015, and October 30, 2015, the licensee provided the sequence of vital loads for each of the following events, if they were to occur while the EDG was operating in parallel with the offsite power source during performance of the above SRs in Mode 1 or 2:

(a) LOOP
(b) LOCA
(c) LOOP concurrent with LOCA
(d) LOOP followed by LOCA
(e) LOCA followed by LOOP

3.4.1 Loss of Offsite Power

In the event of a LOOP while the EDG is operating in parallel with the offsite power source, the EDG will continue to supply all 4.16 KV safety and BOP buses. Since the EDG remains connected to the 4.16 KV buses, the LOV relays cannot sense the LOOP event at the onset of grid loss. All connected BOP loads in excess of 15 megavolts amperes (MVA), originally supplied by the RATs, will overload the EDG resulting in an overcurrent trip of the EDG. The overcurrent relay actuation will actuate the EDG lockout relay, which will require manual reset at the control room panel. Once the EDG trips on overcurrent, the LOV relays will sense the loss of voltage on the 4.16 KV safety buses and generate a LOOP signal. Load shedding will be initiated and all 4.16 KV loads and 600 V block loads will shed. The LOOP signal will trip open the offsite power supply breaker in less than 3 cycles. Once the lockout relays are manually reset, the EDG will restart in the emergency mode. After the EDG reaches rated voltage and frequency, the EDG output breaker will re-close onto the 4.16 KV safety bus, and the emergency LOOP loads will automatically re-sequence onto the bus with sequence timers as follows: the component cooling water (CCW) pump at 13.16 seconds, essential service water (ESW) pump at 16.65 seconds, motor driven auxiliary feed water (AFW) pump at 20.51 seconds, and the non-essential service water (NESW) pump at 24.78 seconds. Timing of the LOOP loads start after the lockout relays are manually reset, and includes 10 seconds for the EDG to start and reach required voltage.

If a sustained degraded voltage condition is detected by the DVR logic, the DVRs actuate auxiliary relays that in turn initiate tripping of the normal supply circuit breakers to the 4.16 KV safety buses. As described above for the LOOP event, when power to the 4.16 KV safety buses is completely lost, the LOV relays will detect the loss of voltage and generate a LOOP signal. Load shed will be initiated and remaining automatic actions will occur as described above.

In addition, the licensee stated that the EDG lockout is the same response that would occur for existing surveillance SR 3.8.1.3, the EDG monthly surveillance test run, which is performed every month when the EDG is paralleled to the safety buses. Therefore, the EDG is declared inoperable during surveillance testing. If the EDG were to lockout due to overloading during the surveillance testing, dedicated operators are positioned at the EDG control panel to manually reset the lockout relay.

3.4.2 Loss-of-Coolant Accident

If a LOCA event occurs while the EDG being tested is paralleled to offsite power, the Safety Injection signal will automatically trip the EDG output breaker to separate the EDG from the 4.16 KV safety bus. The EDG will continue to run unloaded in standby mode. Following a reactor trip, the main generator will trip after 30 seconds, thereafter, the normal source breakers from the UAT to the 4.16 KV BOP buses will trip, and the reserve source breakers from the RATs will close. The 4.16 KV BOP and safety buses will continue to be powered from offsite power via the RATs, engineered safety system loads will be powered from the plant safety buses via the RATs, and the EDG will remain operating in a standby mode. The following emergency loads will be sequenced onto the 4.16 KV safety bus if they were not already running: the centrifugal charging pump, safety injection pump, residual heat removal (RHR) pump, CCW pump, ESW pump, motor driven AFW pump, CS pump, and the NESW pump.

3.4.3 Loss of Offsite Power Concurrent with Loss-of-Coolant Accident

If a LOOP concurrent with LOCA event occurs during EDG testing in parallel with the offsite power source, the Safety Injection signal will trip the output breaker of the EDG, which separates the EDG from the 4.16 KV safety bus. The EDG will continue to run unloaded in standby mode. The LOOP event will result in LOV relays actuating after the 2 second time delay, once the 4.16 KV bus voltage drops below 78 percent. The actuation of LOV relays initiates load shedding and tripping of the tie breakers to the safety-related 4.16 KV buses. After a 2 second time delay, the EDG output breakers will re-close automatically on the 4.16 KV buses, and the emergency motors will be sequenced onto the 4.16 KV safety buses as follows, with time measured from the closing of the EDG output breaker: the centrifugal charging pump at 3.16 seconds, safety injection pump at 6.65 seconds, RHR pump at 10.51 seconds, CCW pump at 14.78 seconds, ESW pump at 19.49 seconds, motor driven AFW pump at 24.69 seconds, CS pump at 30.44 seconds, and the NESW pump at 36.80 seconds.

3.4.4 Loss of Offsite Power Followed by Loss-of-Coolant Accident

Initially during the LOOP, the EDG will remain connected to the 4.16 KV buses, thus LOV relays will not sense a LOOP event at the onset of grid loss. All BOP loads (estimated in excess of 15 MVA), served by offsite power will be picked up by the EDG, resulting in an overcurrent trip of the EDG. The overcurrent relay actuation will lockout the EDG output breaker. After the EDG trip, the power to the 4.16 KV buses will be completely lost, which will actuate LOV relays, and a load shed will be initiated. Once the lockout relay is manually reset by an operator, the EDG will restart, the EDG output breaker will re-close on the 4.16 KV safety bus, and the following emergency LOOP loads will re-sequence onto the bus as follows. Time is measured from the manual reset of the lockout relay, and includes 10 seconds for the EDG to start and reach required voltage: the CCW pump at 13.16 seconds, ESW pump at 16.65 seconds, motor driven AFW pump at 20.51 seconds, and NESW pump at 24.78 seconds.

After the LOOP, a LOCA signal will result in the start of additional emergency pumps as follows. Time is measured from receipt of the Safety Injection signal: the centrifugal charging pump at 3.16 seconds, safety injection pump at 6.65 seconds, and RHR pump at 10.51 seconds.

3.4.5 Loss-of-Coolant Accident Followed by Loss of Offsite Power

For a LOCA followed by a LOOP event, the Safety Injection signal will trip the output breaker of the EDG, separating the EDG from the 4.16 KV bus. The EDG will continue to run unloaded in standby mode. The 4.16 KV BOP and safety buses will be powered from offsite power via the RATs. The emergency loads will be sequenced onto the 4.16 KV safety buses as follows, with time measured from the receipt of the Safety Injection signal: the centrifugal charging pump at 3.16 seconds, safety injection pump at 6.65 seconds, RHR pump at 10.51 seconds, CCW pump at 14.78 seconds, ESW pump at 19.49 seconds, motor driven AFW pump at 24.69 seconds, and NESW pump at 36.80 seconds. If offsite power is lost during the LOCA event, power to the 4.16 KV buses will be lost. The LOV relays will actuate after a 2 second time delay once the bus voltage drops below 78 percent. The LOV relay actuation will initiate load shedding and tripping of the tie breakers to the 4.16 KV safety buses. The EDG output breakers will re-close on the 4.16 KV safety bus, and the emergency motors will be re-sequenced onto the bus as follows. Time is measured from the closure of the EDG output breaker: the centrifugal charging pump 3.16 seconds, safety injection pump 6.65 seconds, RHR pump 10.51 seconds, CCW pump at 14.78 seconds, ESW pump at 19.49 seconds, motor driven AFW pump at 24.69 seconds, and NESW pump at 36.80 seconds.

3.4.6 Emergency Conditions Conclusion

The NRC staff reviewed the above information provided by the licensee, regarding various combinations of LOOP and LOCA events, if they were to occur during the performance of EDG SRs in Mode 1 or 2. It is expected that the plant response to the LOOP event, and the LOOP followed by LOCA event, will be potentially delayed, as operator action is required to reset the

EDG lockout relay. However, the time response to the LOOP event is not considered critical to safety, therefore a potential delay is acceptable. The staff finds that the plant response to a LOCA event will not be impacted by the EDG testing, since the EDG will continue to run unloaded while the offsite power source supplies the buses. The redundant train of onsite AC power would not be affected by the EDG under test, and thus, would remain available to respond to various design basis LOOP/LOCA events within an appropriate amount of time. Therefore, the NRC staff finds that the performance of SRs 3.8.1.10, 3.8.1.11, and 3.8.1.15 in Modes 1 and 2 will not have an adverse impact on the onsite AC electrical systems, if an emergency condition were to occur.

3.5 Technical Evaluation Conclusion

Based on above evaluations, the staff finds that performing SRs 3.8.1.10, 3.8.1.11, and 3.8.1.15 during Mode 1 or 2 is acceptable due to the following main factors:

- The EDGs will remain adequately protected.
- The onsite AC electrical system will not be adversely affected by the performance of the SRs.
- The licensee currently tests the EDGs paralleled to offsite power or generator output while at power during required monthly surveillance testing, SR 3.8.1.3, the EDG monthly surveillance test run, with no report of adverse effect.
- The SRs will be performed on one EDG at any given time. The other train EDG will remain available and operable to respond to design basis accident conditions.

The NRC staff reviewed the proposed changes to CNP TS SRs 3.8.1.10, 3.8.1.11, and 3.8.1.15 for EDG testing. Performance of these SRs is currently normally prohibited in Mode 1 or 2. The proposed changes would remove the mode restrictions on these SRs to allow the SRs to be performed in any mode. Based on the above technical evaluation, the staff concludes that, with the proposed TS changes, the licensee will continue to comply with the requirements of 10 CFR 50.36(c) and PSDC 39. Therefore, the NRC staff finds that that the EDGs may be safely tested by performing SRs 3.8.1.10, 3.8.1.11, and 3.8.1.15 in Modes 1 and 2, and the proposed changes in the LAR are 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

The 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 and change SRs. The NRC 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 in the *Federal Register* on March 17, 2015 (80 FR 13909) a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding. 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 Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that 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 Contributors: A. Foli, NRR
V. Goel, NRR

Date: December 11, 2015

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

Sincerely,

/RA/

Allison W. Dietrich, Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosures:
1. Amendment No. 330 to DPR-58
2. Amendment No. 311 to DPR-74
3. Safety Evaluation

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