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

March 28, 2017

Ms. Tanya Hamilton
Site Vice President
Shearon Harris Nuclear Power Plant
Duke Energy
5413 Shearon Harris Road
New Hill, NC 27562-0165

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 – 2ND CORRECTION TO LICENSE AMENDMENT NO. 154 REGARDING RISK-INFORMED JUSTIFICATIONS FOR THE RELOCATION OF SPECIFIC SURVEILLANCE FREQUENCY REQUIREMENTS TO A LICENSEE-CONTROLLED PROGRAM (CAC NO. MF6583)

Dear Ms. Hamilton:

By letter dated November 29, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16200A285), the U.S. Nuclear Regulatory Commission (NRC) issued Amendment No. 154 to Renewed Facility Operating License No. NPF-63 for the Shearon Harris Nuclear Power Plant, Unit 1 (Harris). The amendment revised the technical specifications (TSs) by relocating specfic surveillance frequencies to a licensee-controlled program in response to the application from Duke Energy Progress, LLC (Duke Energy) (previously Duke Energy Progress, Inc.) dated August 18, 2015, as supplemented by letters dated September 29, 2015, and February 5, April 28, and May 19, 2016.

On February 21, 2017, the NRC staff issued a correction letter (ADAMS Accession No. ML17047A017) to Amendment 154. Based on subsequent discussions with Duke Energy staff, the NRC staff was informed of several additional errors introduced by Amendment 154 as follows:

TS Page 3/4 3-11, Table 4.3-1, Reactor Trip System Instrumentation Surveillance Requirements:

Functional Unit 1. Manual Reactor Trip

- Missing periods after N.A under CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST
- Missing asterisk on Mode 5 under MODES FOR WHICH SURVEILLANCE IS REQUIRED

Functional Unit 2. Power Range, Neutron Flux, b. Low Setpoint

- Missing periods after N.A under TRIP ACTUATING DEVICE OPERATIONAL TEST and ACTUATION LOGIC TEST

TS Page 3/4 3-12, Table 4.3-1 (continued), Reactor Trip System Instrumentation Surveillance Requirements:

Functional Unit 13. Steam Generator Water Level - - Low-Low

- Missing hyphen in "Low-Low" for FUNCTIONAL UNIT 13. Steam Generator Water Level - - Low-Low

TS Page 3/4 3-17, TS 3/4.3.2, Engineered Safety Features Actuation System Instrumentation:

- Missing period at the end of SR 4.3.2.2.

TS Page 3/4 3-41, Table 4.3-2, Engineered Safety Features Actuation System Instrumentation Surveillance Requirements

Channel Functional Unit 1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Ventilation Isolation, Phase A Containment Isolation, Start Auxiliary Feedwater System Motor-Driven Pumps, Start Containment Fan Coolers, Start Emergency Service Water Pumps, Start Emergency Service Water Booster Pumps), c. Containment Pressure

- Missing hyphen in "High-1" for CHANNEL FUNCTIONAL UNIT 1.c. Containment Pressure – High-1

TS Page 3/4 3-45, Table 4.3-2 (continued), Engineered Safety Features Actuation System Instrumentation Surveillance Requirements

Channel Functional Unit 5. Turbine Trip and Feedwater Isolation, a. Automatic Actuation Logic and Actuation Relays

- Missing period after N.A under ANALOG CHANNEL OPERATIONAL TEST

Channel Functional Unit 6. Auxiliary Feedwater, e. Loss-of-Offsite Power Start Motor-Driven Pumps and Turbine Driven Pump

- Missing hyphens in "Loss-of-Offsite" for note "See Item 9. below for all Loss-of-Offsite Power Surveillance Requirements"

TS Page 3/4 4-6, Reactor Coolant System Cold Shutdown - Loops Filled

- Missing period at the end of SR 4.4.1.4.1.1
- Change bars for SR 4.4.1.4.1.1 and SR 4.4.1.4.1.3 do not fully reflect changes made in Amendment 154

TS Page 3/4 4-34, Reactor Coolant System, TS 3/4.4.9 Pressure/Temperature Limits

- Missing hyphen in "1-hour" in LCO 3.4.9.2.c
- Change bar for SR 4.4.9.2.1 does not fully reflect changes made in Amendment 154

TS Page 3/4 5-1, 3/4.5 Emergency Core Cooling Systems, TS 3/4.5.1 Accumulators, Cold Leg Injection

- Missing hyphen in "cover-pressure" in LCO 3.5.1.d

TS Page 3/4 5-9, Emergency Core Cooling Systems, TS 3/4.5.4 Refueling Water Storage Tank

- Stray underscore between "the" and "RWST" in ACTION

TS Page 3/4 7-15, Plant Systems, TS 3/4.7.6 Control Room Emergency Filtration System

- Stray comma following the "or" after the semicolon in ACTION c.1 for LCO 3.7.6
- Missing "the" before "recirculation" in ACTION c.2 for LCO 3.7.6
- Commas inadvertently switched to periods following "C.5.a" and "C.5.c" in SR 4.7.6.b.1

TS Page 3/4 7-31, Plant Systems, 3/4.7.14 Fuel Storage Pool Boron Concentration
- Missing Section Title, "PLANT SYSTEMS"

TS Page 3/4 8-20, Electrical Power Systems, Electrical Equipment Protective Devices, Containment Penetration Conductor Overcurrent Protective Devices
- Missing hyphen in "short-time" in SR 4.8.4.1.a.2.

TS Page 3/4 8-39, Electrical Power Systems, Electrical Equipment Protective Devices, Motor-Operated Valves Thermal Overload Protection
- Missing hyphen in "MOTOR-OPERATED" in the title.

The staff notes that while the amendment request included changes on the pages listed above, the changes listed above were not requested.

It appears that these errors were introduced by the camera-ready TS pages provided by Duke Energy prior to issuance of the amendment. The data on the camera ready pages were not consistent with the changes proposed for (1) TS Page 3/4 4-6 as shown in Attachment 1 to Duke Energy's letter dated May 19, 2016 (ADAMS Accession ML16141A048) and (2) the remaining TS pages as shown in Enclosure 3 to Duke Energy's letter dated August 18, 2015 (ADAMS Accession ML15236A254). The corrected pages are enclosed.

If you have any questions, please contact me at (301) 415-6256 or by e-mail at Dennis.Galvin@nrc.gov.

Sincerely,

Dennis J Galvin

Dennis J. Galvin, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-400

Enclosure:
Corrected TS Pages to Renewed
License No. NPF-63

cc w/enclosure: Distribution via Listserv

CORRECTION TO LICENSE AMENDMENT NO. 154

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

RENEWED FACILITY OPERATING LICENSE NO. NPF-63

DOCKET NO. 50-400

Replace the following page 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
3/4 3-11	3/4 3-11
3/4 3-12	3/4 3-12
3/4 3-17	3/4 3-17
3/4 3-41	3/4 3-41
3/4 3-45	3/4 3-45
3/4 4-6	3/4 4-6
3/4 4-34	3/4 4-34
3/4 5-1	3/4 5-1
3/4 5-9	3/4 5-9
3/4 7-15	3/4 7-15
3/4 7-31	3/4 7-31
3/4 8-20	3/4 8-20
3/4 8-39	3/4 8-39

Enclosure

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	SFCP(12)	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux						
a. High Setpoint	SFCP	SFCP (2,4), SFCP (3,4), SFCP (4,6), SFCP (4,5)	SFCP	N.A.	N.A.	1, 2
b. Low Setpoint	SFCP	SFCP (4)	S/U(1)	N.A.	N.A.	1***, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	SFCP (4)	SFCP	N.A.	N.A.	1, 2
4. Power Range, Neutron Flux, High Negative Rate	N.A.	SFCP (4)	SFCP	N.A.	N.A.	1, 2
5. Intermediate Range, Neutron Flux	SFCP	SFCP (4,5)	S/U(1)	N.A.	N.A.	1***, 2
6. Source Range, Neutron Flux	SFCP	SFCP (4,5)	S/U(1), SFCP(8)	N.A.	N.A.	2**, 3, 4, 5
7. Overtemperature ΔT	SFCP	SFCP (11)	SFCP	N.A.	N.A.	1, 2
8. Overpower ΔT	SFCP	SFCP	SFCP	N.A.	N.A.	1, 2
9. Pressurizer Pressure -- Low	SFCP	SFCP	SFCP	N.A.	N.A.	1 (16)
10. Pressurizer Pressure -- High	SFCP	SFCP	SFCP	N.A.	N.A.	1, 2

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
11. Pressurizer Water Level--High	SFCP	SFCP	SFCP	N.A.	N.A.	1
12. Reactor Coolant Flow--Low	SFCP	SFCP	SFCP	N.A.	N.A.	1
13. Steam Generator Water Level--Low-Low	SFCP	SFCP	SFCP(16)	N.A.	N.A.	1, 2 (16)
14. Steam Generator Water Level--Low Coincident with Steam/Feedwater Flow Mismatch	SFCP	SFCP	SFCP	N.A.	N.A.	1, 2
15. Undervoltage -- Reactor Coolant Pumps	N.A.	SFCP	N.A.	SFCP(9)	N.A.	1
16. Underfrequency -- Reactor Coolant Pumps	N.A.	SFCP	N.A.	SFCP(9)	N.A.	1
17. Turbine Trip						
a. Low Fluid Oil Pressure	N.A.	SFCP	N.A.	S/U(1,9)	N.A.	1
b. Turbine Throttle Valve Closure	N.A.	SFCP	N.A.	S/U(1,9)	N.A.	1
18. Safety Injection Input from ESF	N.A.	N.A.	N.A.	SFCP	N.A.	1, 2
19. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	SFCP(4)	SFCP	N.A.	N.A.	2**

SURVEILLANCE REQUIREMENTS

4.3.2.1 Each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays shall be demonstrated OPERABLE by performance of the ESFAS Instrumentation Surveillance Requirements specified in Table 4.3-2.

4.3.2.2 The ENGINEERED SAFETY FEATURES RESPONSE TIME of each ESFAS function shall be verified to be within its limit specified in the Technical Specification Equipment List Program, plant procedure PLP-106, at the frequency specified in the Surveillance Frequency Control Program.

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Ventilation Isolation, Phase A Containment Isolation, Start Auxiliary Feedwater System Motor-Driven Pumps, Start Containment Fan Coolers, Start Emergency Service Water Pumps, Start Emergency Service Water Booster Pumps)								
a. Manual Initiation	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	SFCP(1)	SFCP(3)	1, 2, 3, 4
c. Containment Pressure -- High-1	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
d. Pressurizer Pressure -- Low	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure -- Low	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Main Steam Line Isolation (Continued)								
d. Steam Line Pressure --Low	See Item 1.e. above for Steam Line Pressure --Low Surveillance Requirements.							
e. Negative Steam Line Pressure Rate--High	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	3**, 4**
5. Turbine Trip and Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	SFCP(1)	SFCP	1, 2
b. Steam Generator Water Level--High -High (P-14)	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2
c. Safety Injection	See Item 1. above for Safety Injection Surveillance Requirements.							
6. Auxiliary Feedwater								
a. Manual Initiation	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	SFCP(1)	SFCP	1, 2, 3
c. Steam Generator Water Level--Low-Low	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Safety Injection Start Motor-Driven Pumps	See Item 1. above for all Safety Injection Surveillance Requirements.							
e. Loss-of-Offsite Power Start Motor-Driven Pumps and Turbine-Driven Pump	See Item 9. below for all Loss-of-Offsite Power Surveillance Requirements.							

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation*, and either:

a. One additional RHR loop shall be OPERABLE**, or

b. The secondary side water level of at least two steam generators shall be greater than 74% wide range (WR) or greater than 30% narrow range (NR).

APPLICABILITY: MODE 5 with reactor coolant loops filled***.

ACTION:

a. With one of the RHR loops inoperable and with less than the required steam generator water level, immediately initiate corrective action to return the inoperable RHR loop to OPERABLE status or restore the required steam generator water level as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4.1.1 The secondary side water level of at least two steam generators when required shall be determined to be within limits at the frequency specified in the Surveillance Frequency Control Program.

4.4.1.4.1.2 At least one RHR loop shall be determined to be in operation and circulating reactor coolant at the frequency specified in the Surveillance Frequency Control Program.

4.4.1.4.1.3 Verify required RHR loop locations susceptible to gas accumulation are sufficiently filled with water at the frequency specified in the Surveillance Frequency Control Program.

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* The RHR pump may be deenergized for up to 1 hour provided: (1) no operations are permitted that would cause dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

** One RHR loop may be inoperable for up to 2 hours for surveillance testing provided the other RHR loop is OPERABLE and in operation.

*** A reactor coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 325°F unless the secondary water temperature of each steam generator is less than 50°F above each of the Reactor Coolant System cold leg temperatures.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

LIMITING CONDITION FOR OPERATION

3.4.9.2 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, and inservice leak and hydrostatic testing with:

a. A maximum heatup rate as shown on Table 4.4-6.

b. A maximum cooldown rate as shown on Table 4.4-6.

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: MODES 4, 5, and 6 with reactor vessel head on.

ACTION:

With any of the pressure limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; if the pressure and temperature limit lines shown on Figure 3.4-2 and 3.4-3 were exceeded, perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or maintain the RCS T_{avg} and pressure at less than 200°F and 500 psig, respectively.

SURVEILLANCE REQUIREMENTS

4.4.9.2.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at the frequency specified in the Surveillance Frequency Control Program during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.2.2 Deleted from Technical Specifications. Refer to the Technical Specification Equipment List Program, plant procedure PLP-106.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

COLD LEG INJECTION

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE with:

a. The isolation valve open with power supply circuit breaker open,

b. A contained borated water volume of between 66 and 96% indicated level,

c. A boron concentration of between 2400 and 2600 ppm, and

d. A nitrogen cover-pressure of between 585 and 665 psig.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve or boron concentration not within limits, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

c. With one accumulator inoperable due to boron concentration not within limits, restore the boron concentration within limits within 72 hours or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by:

1. Verifying that the contained borated water volume and nitrogen cover-pressure in the tanks are within their limits, and

2. Verifying that each accumulator isolation valve is open.

*RCS pressure above 1000 psig.

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained borated water volume of 436,000 gallons, which is equivalent to 92% indicated level.

b. A boron concentration of between 2400 and 2600 ppm of boron,

c. A minimum solution temperature of 40°F, and

d. A maximum solution temperature of 125°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour* or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The RWST shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by:

1. Verifying the contained borated water volume in the tank, and

2. Verifying the boron concentration of the water.

b. At the frequency specified in the Surveillance Frequency Control Program by verifying the RWST temperature when the outside air temperature is less than 40°F or greater than 125°F.

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* Except that while performing surveillance 4.4.6.2.2, the tank must be returned to OPERABLE status within 12 hours.

LIMITING CONDITION FOR OPERATION (Continued)

c. During movement of irradiated fuel assemblies or movement of loads over spent fuel pools.

1. With one CREFS train inoperable for reasons other than an inoperable CRE boundary, restore the inoperable CREFS train to OPERABLE status within 7 days or immediately initiate and maintain operation of the remaining OPERABLE CREFS train in the recirculation mode; or immediately suspend movement of irradiated fuel.

2. With both CREFS trains inoperable for reasons other than an inoperable CRE boundary, or with the OPERABLE CREFS train required to be in the recirculation mode by Action c.1., not capable of being powered by an OPERABLE emergency power source, immediately suspend all operations involving movement of irradiated fuel assemblies or movement of loads over spent fuel pools.

3. With one or more CREFS trains inoperable due to inoperable CRE boundary, immediately suspend movement of irradiated fuel assemblies or movement of loads over spent fuel pools.

SURVEILLANCE REQUIREMENTS

4.7.6 Each CREFS train shall be demonstrated OPERABLE:

a. At the frequency specified in the Surveillance Frequency Control Program by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 continuous hours with the heaters operating;

b. At the frequency specified in the Surveillance Frequency Control Program or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following significant painting, fire, or chemical release in any ventilation zone communicating with the system by:

1. Verifying that the cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of less than 0.05% and uses the test procedure guidance in Regulatory Position C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 4000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1980; and

3/4.7.14 FUEL STORAGE POOL BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.7.14 The boron concentration of spent fuel pools shall be ≥ 2000 ppm.

APPLICABILITY: At ALL TIMES for pools that contain nuclear fuel.

ACTION:

a. With the spent fuel pool boron concentration not within the limits, immediately suspend movement of fuel assemblies.

b. Immediately initiate action to restore pool boron concentration within the limit.

SURVEILLANCE REQUIREMENTS

4.7.14 At the frequency specified in the Surveillance Frequency Control Program verify spent fuel pool boron concentration is within the limit by:

a. Sampling the water volume connected to or in applicable pools.

b. In addition to 4.7.14.a, sampling an individual pool containing nuclear fuel if the pool is isolated from other pools.

SURVEILLANCE REQUIREMENTS (Continued)

4.8.4.1 (Continued)

c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current with a value equal to 300% of the pickup of the long-time delay trip element and 150% of the pickup of the short-time delay trip element, and verifying that the circuit breaker operates within the time delay band width for that current specified by the manufacturer. The instantaneous element shall be tested by injecting a current equal to ±20% of the pickup value of the element and verifying that the circuit breaker trips instantaneously with no intentional time delay. Molded case circuit breaker testing shall also follow this procedure except that generally no more than two trip elements, time delay and instantaneous, will be involved. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

b. At the frequency specified in the Surveillance Frequency Control Program by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

ELECTRICAL POWER SYSTEMS
ELECTRICAL EQUIPMENT PROTECTIVE DEVICES
MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

LIMITING CONDITION FOR OPERATION

3.8.4.2 The thermal overload protection of each valve, specified in the Technical Specification Equipment List Program, plant procedure PLP-106, requiring bypass protection, shall be bypassed only under accident conditions by an OPERABLE bypass device.

APPLICABILITY: Whenever the motor-operated valve is required to be OPERABLE.

ACTION:

With the thermal overload protection for one or more of the above required valves not capable of being bypassed under conditions for which it is designed to be bypassed, restore the inoperable device or provide a means to bypass the thermal overload within 8 hours, or declare the affected valve(s) inoperable and apply the appropriate ACTION Statement(s) of the affected system(s).

SURVEILLANCE REQUIREMENTS

4.8.4.2 The thermal overload protection for the above required valves shall be verified to be bypassed only under accident conditions by an OPERABLE integral bypass device by the performance of a TRIP ACTUATION DEVICE OPERATIONAL TEST of the bypass circuitry:

a. At the frequency specified in the Surveillance Frequency Control Program for those thermal overloads which are normally in force during plant operation and are bypassed only under accident conditions; and

b. Following maintenance on the thermal overload bypass relays and circuitry.

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