Tanya M. Hamilton
Vice President
Harris Nuclear Plant
5413 Shearon Harris Road
New Hill, NC 27562-9300

919-362-2502

10 CFR 50.90

July 25, 2019
Serial: RA-19-0123

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Shearon Harris Nuclear Power Plant, Unit 1
Docket No. 50-400
Renewed License No. NPF-63

Subject: License Amendment Request to Eliminate Certain Technical Specification Requirements In Alignment with Improved Standard Technical Specifications

Ladies and Gentlemen:

Pursuant to 10 CFR 50.90, Duke Energy Progress, LLC (Duke Energy), hereby requests a revision to the Technical Specifications (TS) for the Shearon Harris Nuclear Power Plant, Unit 1 (HNP). Specifically, the proposed change would revise TS 3/4.10.3, "Special Test Exceptions, Physics Tests," and TS 3/4.10.4, "Special Test Exceptions, Reactor Coolant Loops," to eliminate the "within 12 hours" restriction from Surveillance Requirement (SR) 4.10.3.2 for performing an Analog Channel Operational Test (ACOT) on the intermediate and power range neutron monitors prior to initiating physics tests and to eliminate the "within 12 hours" restriction from SR 4.10.4.2 for performing an ACOT on the intermediate range monitors, power range monitors, and P-7 interlock prior to initiating startup or physics tests, respectively. These changes are consistent with Technical Specification Task Force (TSTF) Traveler TSTF-108, Revision 1, "Eliminate the 12 hour COT [Channel Operational Test] on power range and intermediate range channels for Physics Test Exceptions" (ADAMS Accession No. ML040480061). The proposed change would also delete certain reporting requirements from TS, provide clarification to required actions of TS 3/4.4.4, "Relief Valves," eliminate a second Completion Time associated with TS 3/4.8.1.1, "A.C. Sources – Operating," and reflect the title changes of the Plant Nuclear Safety Committee (PNSC) to the On-Site Review Committee (ORC) and the Plant General Manager to the plant manager. The proposed change to eliminate a second Completion Time in TS 3/4.8.1.1 is aligned with the justification provided in NRC-approved TSTF-439, "Eliminate Second Completion Times Limiting Time From Discovery of Failure to Meet an LCO" (ADAMS Accession No. ML051860296). These changes are consistent with Revision 4 of NUREG-1431, "Standard Technical Specifications – Westinghouse Plants" (ADAMS Accession No. ML12100A222).

The proposed changes have been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c), and it has been concluded that the proposed changes involve no significant hazards consideration. Enclosure 1 of this license amendment request provides Duke Energy's evaluation of the proposed changes. Enclosure 2 provides a copy of the proposed TS changes. Enclosure 3 provides a copy of the proposed TS Bases changes for information only, as they will be implemented in accordance with the TS Bases Control Program upon implementation of the amendment.

Approval of the proposed license amendment is requested within twelve months of acceptance. The amendment shall be implemented within 90 days from approval.

In accordance with 10 CFR 50.91, a copy of this application, with enclosures, is being provided to the designated North Carolina State Official.

This document contains no new Regulatory Commitments.

Please refer any questions regarding this submittal to Art Zaremba, Manager – Nuclear Fleet Licensing, at (980) 373-2062.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on July 25, 2019.

Sincerely,

Tanya M. Hamilton

Tanya M. Hamilton

Enclosures:

1. Evaluation of the Proposed Changes
2. Proposed Technical Specification Changes
3. Technical Specification Bases Changes

cc: J. Zeiler, NRC Sr. Resident Inspector, HNP
W. L. Cox, III, Section Chief, N.C. DHSR
M. Barillas, NRC Project Manager, HNP
NRC Regional Administrator, Region II

SERIAL RA-19-0123

ENCLOSURE 1

EVALUATION OF THE PROPOSED CHANGES

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

RENEWED LICENSE NUMBER NPF-63

25 PAGES PLUS THE COVER

Evaluation of the Proposed Changes
License Amendment Request to Eliminate Certain Technical Specification Requirements In Alignment with Improved Standard Technical Specifications

1.0 Summary Description

Pursuant to 10 CFR 50.90, Duke Energy Progress, LLC (Duke Energy), hereby requests a revision to the Technical Specifications (TS) for the Shearon Harris Nuclear Power Plant, Unit 1 (HNP). Specifically, the proposed change would revise TS 3/4.10.3, “Special Test Exceptions, Physics Tests,” and TS 3/4.10.4, “Special Test Exceptions, Reactor Coolant Loops,” to eliminate the “within 12 hours” restriction from Surveillance Requirement (SR) 4.10.3.2 for performing an Analog Channel Operational Test (ACOT) on the intermediate and power range neutron monitors prior to initiating physics tests and to eliminate the “within 12 hours” restriction from SR 4.10.4.2 for performing an ACOT on the intermediate range monitors, power range monitors, and P-7 interlock prior to initiating startup or physics tests, respectively. These changes are consistent with Technical Specification Task Force (TSTF) Traveler TSTF-108, Revision 1, “Eliminate the 12 hour COT [Channel Operational Test] on power range and intermediate range channels for Physics Test Exceptions” (Reference 6.2). The proposed change would also delete certain reporting requirements from TS, provide clarification to required actions of TS 3/4.4.4, “Relief Valves,” eliminate a second Completion Time associated with TS 3/4.8.1.1, “A.C. Sources – Operating,” and reflect the title changes of the Plant Nuclear Safety Committee (PNSC) to the On-Site Review Committee (ORC) and the Plant General Manager to the plant manager. The proposed change to eliminate a second Completion Time in TS 3/4.8.1.1 is aligned with the justification provided in NRC-approved TSTF-439, “Eliminate Second Completion Times Limiting Time From Discovery of Failure to Meet an LCO [Limiting Condition for Operation]” (Reference 6.3). These changes are consistent with Revision 4 of NUREG-1431, “Standard Technical Specifications – Westinghouse Plants” (Reference 6.1).

2.0 Detailed Description

2.1 Background

System Design and Operation: Nuclear Instrumentation System

The Excore Nuclear Instrumentation System (NIS) has a safety function to protect the reactor by monitoring the neutron flux level and generating appropriate trips to the Reactor Trip System (RTS). The NIS will initiate reactor trip signals or alarms if preset limits are exceeded indicating unsafe conditions during various phases of reactor operations and during shutdown conditions at all times. The NIS uses three types of instrumentation channels to provide three discrete, overlapping protection levels: Source, Intermediate and Power. Each range provides necessary overpower reactor trip protection required during operation in that range, with permissive conditions that overlap to provide continuous protection. The three types of detectors indicated in the excore NIS can monitor neutron flux leakage from completely shutdown to 120 percent of full power.

A number of permissives and interlocks are associated with the RTS / Engineering Safety Feature Actuation System (ESFAS). These interlocks allow the operator or a system to perform a function only when certain specific conditions have been met.

Permissive P-7 impacts the “at power” trips and allows for orderly shutdown and startup operations without a reactor trip. It is switched by 3/4 power range nuclear instruments above setpoint and 2/2 turbine impulse chamber pressure channels (P-13) signals below setpoint. The trip functions are restored when 2/4 NIS power range (P-10) or 1/2 turbine impulse chamber pressure signals exceed the setpoint.

System Design and Operation: Power-Operated Relief Valves

In MODES 1, 2, and 3, the power-operated relief valves (PORVs) provide a pressure boundary for the Reactor Coolant System (RCS) and a manual RCS pressure control for mitigation of accidents, both of which are safety-related functions. They additionally provide an automatic RCS pressure relief to minimize challenges to the safety valves. The capability of the PORV to provide an RCS pressure boundary requires that the PORV, or its associated block valve, be closed. The capability of the PORV to perform manual RCS pressure control for mitigation of a Steam Generator Tube Rupture accident is based on manual actuation and does not require the automatic RCS pressure control function. As such, the automatic RCS pressure control function of the PORVs is not a safety-related function. While it limits the number of challenges to the safety valves, the safety valves perform the safety function of RCS overpressure protection. Consequently, the automatic RCS pressure control function of the PORVs is not required to be available for the PORVs to be operable.

Second Completion Times

Early revisions of NUREG-1431 included a second Completion Time for certain Required Actions as a means to establish a limit on the maximum time allowed for any combination of Conditions that result in a single continuous failure to meet the LCO. In providing a limit on the amount of time that the LCO could not be met for various combinations of Conditions, these Completion Times (henceforth referred to as "second Completion Times") were intended to preclude entry into and out of the Actions for an indefinite period of time without meeting the LCO. By letter dated November 4, 1994 (Reference 6.4), the NRC issued Amendment No. 51 to the HNP operating license to revise TS 3/4.8.1, “A.C. Sources,” to be consistent with NUREG-1431, Rev. 0. This revision included the addition of Action g to HNP TS 3/4.8.1.1 regarding contiguous events of either an offsite or onsite A.C. source becoming inoperable and resulting in failure to meet the LCO.

Revision 2 of TSTF-439, as approved by the NRC and implemented in Revision 3.1 of NUREG-1431 per Reference 6.5, deletes these second Completion Times from the affected Required Actions. It also established the licensee requirement to have administrative controls to limit the maximum time allowed for any combination of Conditions that result in a single contiguous occurrence of failing to meet the LCO. These administrative controls are to ensure that the maximum time allowed for any combination of Conditions that result in a single contiguous occurrence of failing to meet the LCO is not inappropriately extended.

Special Reports

Regulatory Guide (RG) 1.16, Revision 1, "Reporting of Operating Information," was published by the NRC in October 1973 to provide an acceptable basis for meeting the reporting requirements of the facility operating license. It provided a description of each of the periodic reports, including annual reports and the Startup Report, that licensees are required to submit to demonstrate compliance with the TS reporting requirements. The NRC withdrew RG 1.16 in August 2009 via the *Federal Register* (74 FR 40244) on the basis that it was no longer needed since TS reporting requirements are contained in 10 CFR 50, as well as other parts of 10 CFR Chapter 1.

Additionally, the results of an NRC information gathering assessment that were provided in Generic Letter (GL) 97-02, "Revised Contents of the Monthly Operating Report," identified the existence of duplicative reporting and determined that some reports could be reduced in scope or eliminated.

2.2 Current Technical Specification Requirements

The HNP TS are based upon the format and content of the NUREG-0452, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors," series (Reference 6.6). As a result, the HNP TS surveillance numbers and associated Bases numbers differ from those contained in NUREG-1431 (Reference 6.1).

HNP TS contain special test exceptions to permit relaxation of existing LCO requirements to allow startup and to allow certain physics tests to be performed, as described below:

- TS 3/4.10.3, Special Test Exceptions, Physics Tests: The limitations of Specifications 3.1.1.3, 3.1.1.4, 3.1.3.1, 3.1.3.5, and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

    a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,
    b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set at less than or equal to 25% of RATED THERMAL POWER, and
    c. The Reactor Coolant System lowest operating loop temperature (T_{avg}) is greater than or equal to 541°F.

    This TS is applicable to Mode 2 Operation. It is designed to permit certain physics tests to be performed at less than or equal to 5% of rated thermal power with the Reactor Coolant System T_{avg} slightly lower than normally allowed so that the fundamental nuclear characteristics of the core and related instrumentation can be verified. For various characteristics to be accurately measured, it is at times necessary to operate outside the normal restrictions of these TS.

    SR 4.10.3.2 requires, “Each Intermediate and Power Range channel shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.” *[Emphasis added.]*

- TS 3/4.10.4, Special Test Exceptions, Reactor Coolant Loops: The limitations of Specification 3.4.1.1 may be suspended during the performance of startup and PHYSICS TESTS in MODE 1 or 2 provided:

    a. The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and
    b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set less than or equal to 25% of RATED THERMAL POWER.

This special test exception permits reactor criticality under no flow conditions and is required to perform certain startup and physics tests while at low thermal power levels.

SR 4.10.4.2 requires “Each Intermediate and Power Range channel, and P-7 Interlock shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST <u>within 12 hours</u> prior to initiating startup and PHYSICS TESTS.” *[Emphasis added.]*

HNP TS also contain special reporting requirements associated with various TS actions, surveillance requirements, fuel assembly design features, and administrative controls. These include:

- TS 3/4.1.1.3, Reactivity Control Systems, Moderator Temperature Coefficient: With the MTC more positive than the Positive MTC Limit specified in the COLR, a special report is required to be prepared and submitted to the NRC within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.
- TS 3/4.4.9.4, Reactor Coolant System, Overpressure Protection System: In the event either the PORVs or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the NRC within 30 days, describing the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.
- TS 3/4.5.2, Emergency Core Cooling Systems, ECCS Subsystems – T_{avg} Greater Than Or Equal To 350°F: In the event the ECCS is actuated and injects water into the RCS, a Special Report shall be prepared and submitted to the NRC within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. Additionally, the current values of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.
- TS 3/4.5.3, Emergency Core Cooling Systems, ECCS Subsystems – T_{avg} Less Than 350°F: In the event the ECCS is actuated and injects water into the RCS, a Special Report shall be prepared and submitted to the NRC within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. Additionally, the current values of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.
- TS 3/4.6.1.6, Containment Systems, Containment Vessel Structural Integrity: SR 4.6.1.6.2 requires that any abnormal degradation of the containment vessel structure detected during required inspections shall be reported to the NRC in a Special Report within 15 days, describing the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.
- TS 3/4.7.9, Plant Systems, Sealed Source Contamination: SR 4.7.9.3 requires that a report be prepared and submitted to the NRC on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microCurie of removable contamination.
- TS 5.3.1, Reactor Core, Fuel Assemblies: Should more than a total of 30 fuel rods or more than 10 fuel rods in any one assembly be replaced per refueling, a Special Report is required to be submitted to the NRC within 30 days after cycle startup describing the number of rods replaced.
- TS 6.9.1.1, Startup Report: A summary report of plant startup and power escalation testing is required to be submitted to the NRC following: (1) receipt of an Operating

License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit. This report shall address each of the tests identified in the Final Safety Analysis Report and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described, along with any additional specific details required in license conditions based on other commitments. This report is to be submitted within: (1) 90 days following completion of the Startup Test Program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality. If the report does not cover all three events (i.e., initial criticality, completion of Startup Test Program, and resumption or commencement of commercial operation), supplementary reports are to be submitted at least every 3 months until all three events have been completed.

- TS 6.9.1.2, Annual Reports: Prior to March 1 of each year, an annual report is required to be submitted to the NRC covering the results of specific activity analyses in which the reactor coolant exceeded the TS limits. The report includes the following information: (1) reactor power history starting 48 hours prior to the first sample in which the limit was exceeded (in graphic and tabular format); (2) results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit, along with the date and time of sampling and the radioiodine concentrations; (3) cleanup flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) graph of the I-131 concentration (μCi/gm) as a function of time for the duration of the specific activity above the steady-state level; and (5) the time duration when the specific activity of the reactor coolant exceeded the radioiodine limit.

The HNP TS for Reactor Coolant System relief valves, TS 3/4.4.4, requires that all PORVs and their associated block valves be operable. In the event that one or more block valves are inoperable, ACTION c of the TS directs the licensee to: 1) restore the block valve(s) to operable status, or close the block valve(s) and remove power from the block valve(s), or close the PORV and remove power from its associated solenoid valve; and 2) apply the ACTION b, as appropriate, for the isolated PORV(s). ACTION b of the TS provides the actions and time requirements for one or more inoperable PORVs as follows:

b. With one or more PORV(s) inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s), and
    1. With only one safety grade PORV OPERABLE, restore at least a total of two safety grade PORVs to OPERABLE status within the following 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or
    2. With no safety grade PORVs OPERABLE, restore at least one safety grade PORV to OPERABLE status within 1 hour and follow ACTION b.1, above, with the time requirement of that ACTION statement based on the time of initial loss of the remaining inoperable safety grade PORV or be in at least HOT

STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

ACTION g of TS 3.8.1.1, A.C. Sources – Operating, requires a licensee with contiguous events of either an offsite or onsite A.C. source becoming inoperable and resulting in failure to meet the LCO to restore all A.C. sources required by 3.8.1.1 within 6 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Administrative Control TS 6.1, Responsibility, identifies the Plant General Manager as the individual responsible for overall unit operation. It also requires the Plant General Manager to delegate in writing the succession to this responsibility during his absence.

Administrative Control TS 6.6, Reportable Event Action, requires that each reportable event be reviewed by the PNSC, with the results of the review submitted to the Nuclear Assessment Section Manager and the site Vice President.

Administrative Control TSs 6.13, Process Control Program (PCP), and 6.14, Offsite Dose Calculation Manual (ODCM), require review and acceptance by the PNSC and approval of the Plant General Manager prior to changes to the PCP or ODCM becoming effective.

2.3 Reason for Proposed Changes

As stated above, HNP TS are based upon the format and content of NUREG-0452. However, the NRC allows for selective incorporation of Improved Standard Technical Specifications (STS) requirements (i.e., NUREG-1431 for Westinghouse Plants - Reference 6.1). As discussed in Section 16.0 of NUREG-0800 (Reference 6.7), TS change requests for facilities with TS based on previous STS should comply with comparable provisions in current STS NUREGs to the extent possible or justify deviations from the STS. The proposed changes found in this license amendment request are generally consistent with the requirements in the current STS.

Changes associated with redundant testing

The performance of an ACOT to verify the intermediate and power range nuclear instrument channels and P-7 are already required per LCO 3.3.1, Reactor Trip System Instrumentation. SR 4.3.1.1 requires “Each Reactor Trip System instrumentation channels and interlock and the automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirements specified in Table 4.3-1.”

- For Power Range monitors – high setpoint, Table 4.3-1, Item 2a, specifies an ACOT at a frequency as determined through the Surveillance Frequency Control Program, while the unit is in Modes 1 or 2.
- For Power Range monitors – low setpoint, Table 4.3-1, Item 2b, specifies an ACOT prior to reactor startup, if not performed in the previous 31 days, while the unit is in Modes 1 or 2 while below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.
- For the Intermediate Range monitors, Table 4.3-1, Item 5, specifies an ACOT prior to reactor startup, if not performed in the previous 31 days, while the unit is in Modes 1 or 2 while below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

- For the Reactor Trip System Interlocks – Low Power Reactor Trips Block, P-7, Table 4.3-1, Item 19b, specifies an ACOT at a frequency as determined through the Surveillance Frequency Control Program, while the unit is in Mode 1.

SR 4.10.3.2 and SR 4.10.4.2 require an additional ACOT performed 12 hours prior to startup or physics testing regardless of whether the above ACOT has been performed within the specified frequency. This is an extraneous and unnecessary performance of a surveillance.

Changes associated with deletion of special reports

The intent of the proposed change is to eliminate redundant reports that are no longer considered warranted, as consistent with the *Federal Register* Notice that withdrew RG 1.16, the results provided in GL 97-02, and the guidance provided in NUREG-1431. Reporting would continue to be based on the NRC's regulatory requirements as prescribed in 10 CFR Part 50.

Changes associated with providing additional clarification

In its current state, TS 3/4.4.4 ACTION c instructs the licensee to apply ACTION b, as appropriate, for the isolated PORV(s) resulting from the inoperable block valve(s). Clarification is needed for ACTION c to identify which parts of ACTION b are appropriate whenever the PORV is declared inoperable due to inoperable block valve(s).

Changes associated with removal of second completion times

Due to the remote likelihood of experiencing concurrent failures, the inclusion of the second Completion Time in TS 3.8.1.1 did not originally create an operational restriction for HNP. However, it has since become a problem as it relates to future efforts to pursue a risk-informed Completion Time for this Specification. The second Completion Time was intended to preclude entry into and out of the ACTIONS for an indefinite period of time without meeting the LCO as achieved by providing a limit on the amount of time that the LCO could not be met for various combinations of Conditions. In its current state, HNP would not be able to fully utilize the benefits gained with a risk-informed Completion Time until the second Completion Time is eliminated.

Changes associated with renaming of review committee

Duke Energy has moved to a fleet model regarding the development of a multi-disciplined committee responsible for review of activities that have the potential to affect nuclear safety. The ORC replaces the previously established PNSC, maintaining the responsibility of conducting independent cross-functional review of items related to nuclear safety, safe operation and overall performance and advising the plant manager on these matters.

Changes associated with renaming of plant manager

The title Plant General Manager was a plant-specific position title that has since been updated to reflect a move by Duke Energy to a fleet-wide naming convention. There is no change to, or reassignment of, the responsibility for overall operation of the unit. The generic title “plant manager” is consistent with NUREG-1431.

2.1.4 <u>Description of Proposed Changes</u>

Duke Energy is proposing the deletion of the restriction of "within 12 hours" from both SR 4.10.3.2 and SR 4.10.4.2. This will require the ACOTs to be performed if they are not current (e.g., have not already been performed in accordance with SR 4.3.1.1 required frequencies).

The specific changes are as follows:

- SR 4.10.3.2: "Each Intermediate and Power Range channel shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST ~~within 12 hours~~ prior to initiating PHYSICS TESTS."

- SR 4.10.4.2: "Each Intermediate and Power Range channel, and P-7 Interlock shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST ~~within 12 hours~~ prior to initiating startup and PHYSICS TESTS."

As it relates to the elimination of redundant reporting requirements, Duke Energy is proposing the following changes to TS:

- *TS Index*

  The TS index is revised to reflect the proposed changes in the LAR. These conforming changes to the index are administrative in nature and require no further justification.

- *TS 3/4.1.1.3 – Moderator Temperature Coefficient (MTC)*

  Delete ACTION a.3 requiring a special report:

  ~~3. A Special Report is prepared and submitted to the Commission, pursuant to Specification 6.9.2, within 10 days, describing the value of the measured MRC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.~~

- *TS 3/4.4.9.4 – Overpressure Protection Systems*

  Delete ACTION d requiring a special report:

  ~~d. In the event either the PORVs or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.~~

  Re-letter ACTION e:

  ~~e.~~ **d.** The provisions of Specification 3.0.4 are not applicable.

- *TS 3/4.5.2 – ECCS Subsystems – T_{avg} Greater Than Or Equal To 350°F*

  Delete ACTION b requiring a special report:

  > ~~b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.~~

- *TS 3/4.5.3 – ECCS Subsystems – T_{avg} Less Than 350°F*

  Delete ACTION c requiring a special report:

  > ~~c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.~~

- *TS 3/4.6.1.6 – Containment Vessel Structural Integrity*

  Delete SR 4.6.1.6.2 requiring a special report:

  > ~~4.6.1.6.2 Reports. Any abnormal degradation of the containment vessel structure detected during the above required inspections shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2 within 15 days. This report shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.~~

- *TS 3/4.7.9 – Sealed Source Contamination*

  Delete SR 4.7.9.3 requiring a special report:

  > ~~4.7.9.3 Reports – A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microCurie of removable contamination.~~

- *TS 5.3.1 – Fuel Assemblies*

  Delete special report from TS 5.3.1:

  > 5.3.1 The core shall contain 157 fuel assemblies with each fuel assembly normally containing 264 fuel rods clad with Zircaloy-4 of M5. Limited substitution of fuel rods by filler rods (consisting of Zircaloy-4 or M5 clad

stainless steel or zirconium), or vacancies may be made in fuel assemblies if justified by a cycle specific evaluation. ~~Should more than a total of 30 fuel rods or more than 10 fuel rods in any one assembly be replaced per refueling, a Special Report describing the number of rods replaced will be submitted to the Commission, pursuant to Specification 6.9.2, within 30 days after cycle startup.~~ Each fuel rod shall have a nominal active fuel length of 144 inches. The initial core loading shall have a maximum enrichment of 3.5 weigh percent U 235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 5.0 weight percent U-235.

- *TS 6.9 – Reporting Requirements*

  Delete TS 6.9.1.1, Startup Report.

  Delete TS 6.9.1.2, Annual Reports.

As it relates to providing additional clarification to TS 3/4.4.4 ACTION c, Duke Energy proposes the following change to reflect which selections of ACTION b are appropriate for isolated PORV(s) when one or more block valve(s) are inoperable:

c. With one or more block valve(s) inoperable, within 1 hour: (1) restore the block valve(s) to OPERABLE status, or close the block valve(s) and remove power from the block valve(s), or close the PORV and remove power from its associated solenoid valve, and (2) apply ~~the~~ ACTION ~~b,~~ **b.1 or b.2**, above, as appropriate, for the isolated PORV(s).

Duke Energy also proposes the deletion of TS 3.8.1.1 ACTION g in its entirety, as aligned with the guidance provided in NRC-approved TSTF-439 for deletion of second Completion Times. HNP TSs do not contain a Use and Application Section, nor do they contain second Completion Times for the other TS identified in TSTF-439. Accordingly, there are no other changes associated with implementation of this TSTF.

The TS Bases associated with TS 3.8.1.1 do not contain information related to the second Completion Time. However, they will be revised to incorporate discussion that is similar to and reflects the intent of the guidance that is provided in TSTF-439 relative to proper TS Action entry when the LCO is not met, as follows:

It is possible to alternate between Technical Specification Conditions in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. However, doing so would be inconsistent with the basis for Completion Times. Therefore, the maximum time allowed for any combination of Conditions that result in a single contiguous occurrence of failing to meet the LCO shall be limited. The Completion Times for those Conditions shall not be inappropriately extended.

The proposed changes to Administrative Control TS 6.1, 6.6, 6.13, and 6.14 replace references to the PNSC with the ORC and/or Plant General Manager with plant manager. These changes are not technical in nature, do not modify the responsibility of the review committee as provided in the TS, and do not change or reassign the responsibility for overall operation of the unit. For the actions requiring approval, the plant manager maintains this responsibility, as aligned with NUREG-1431 STS. As such, these proposed changes implement administrative non-technical changes and, given the above, additional technical evaluation of these administrative non-technical changes is not necessary.

3.0 Technical Evaluation

Removal of Redundant Testing Requirements

TS 3/4.10.3.2 – Physics Tests and TS 3/4.10.4.2 – Reactor Coolant Loops

The redundant testing required by SRs 4.10.3.2 and 4.10.4.2, the respective equivalents to NUREG-1431 STS SR 3.1.10.1 and 3.4.19.2, is addressed by TSTF-108-A, as approved by NRC on May 2, 1997. The result of TSTF-108-A is the removal of the 12-hour requirement so that the testing performed for SR 4.3.1.1 may be used to satisfy SRs 4.10.3.2 and 4.10.4.2.

HNP SR 4.10.3.2 and SR 4.10.4.2 require an ACOT "within 12 hours prior" to initiation of startup or physics tests regardless of whether the ACOT has been performed within its required frequency for RTS. Performance of an ACOT on power range and intermediate range channels, as well as the P-7 interlock, is required by TS 3/4.3.1, RTS Instrumentation. Initiation of startup or physics tests does not impact the ability of the monitors to perform their required function, does not affect the trip setpoints or RTS trip capability, and does not invalidate previous surveillances. Therefore, an additional surveillance required to be performed "prior to" these events is an extraneous and unnecessary performance of a surveillance. With the deletion of the restriction of “within 12 hours” from both SR 4.10.3.2 and SR 4.10.4.2, the ACOTs will still be required to be performed if they are not current relative to SR 4.3.1.1 required frequencies.

The elimination of the redundant surveillance requirements does not diminish the required level of testing for the impacted monitors and interlock. The monitors and interlock will continue to be tested at appropriate frequencies by other SRs, within the intervals that have been accepted for HNP. The following testing will continue to be performed:

- SR 4.3.1.1 requires an ANALOG CHANNEL OPERATIONAL TEST be performed on the power range monitors at the frequency specified by the Surveillance Frequency Control Program for the high setpoint and prior to each reactor startup for the low setpoint (if not performed within the previous 31 days) while the unit is in MODES 1 or 2 (TS Table 4.3-1, Functional Unit 2).
- SR 4.3.1.1 requires an ANALOG CHANNEL OPERATIONAL TEST be performed on the intermediate range monitors prior to each reactor startup (if not performed in previous 31 days) while the unit is in MODES 1 or 2 (TS Table 4.3- 1, Functional Unit 5).
- SR 4.3.1.1 requires an ANALOG CHANNEL OPERATIONAL TEST be performed on the P-7 interlock at the frequency specified by the Surveillance Frequency Control Program while the unit is in MODE 1 (TS Table 4.3-1, Functional Unit 19.b).

The proposed changes to SR 4.10.3.2 and SR 4.10.4.2 are aligned with the NRC-approved changes to NUREG-1431 STS as outlined in TSTF-108-A.

Removal of Extraneous Reporting Requirements

TS 3/4.1.1.3 – Moderator Temperature Coefficient (MTC)

Currently, HNP TS 3/4.1.1.3 requires submitting a special report to the NRC if the MTC exceeds its positive limit as specified in the COLR. This report would describe the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition. As part of this license amendment request, Duke Energy is proposing the elimination of this reporting requirement, as consistent with NUREG-1431 STS 3.1.3, “Moderator Temperature Coefficient (MTC),” which does not require submitting a special report.

The report itself is not needed to ensure operation of the facility in a safe manner. The Action associated with HNP TS 3/4.1.1.3 ensures continued safe operation by establishing and maintaining measures that ensure the MTC is restored to within limits. Duke Energy is not proposing any changes other than removing the requirement to submit a special report, as it only provides information. It neither seeks approval from the NRC nor ensures safe operation of the facility during or after the ten days provided to submit the report. As such, elimination of the report is appropriate on the basis that the change is consistent with NUREG-1431 and the report is not necessary to ensure operation of the facility in a safe manner.

TS 3/4.4.9.4 – Overpressure Protection Systems

HNP TS 3/4.4.9.4 currently requires submitting a special report should either the PORVs or the RCS vent(s) are used to mitigate a RCS pressure transient during operation with RCS temperature 325°F or less. The report is required to describe the circumstances initiating the transient, the effects of the PORVs or RCS vent(s) on the transient, and any corrective actions necessary to prevent recurrence. Duke Energy is proposing the elimination of this reporting requirement.

Licensees were previously required to include documentation of challenges to PORVs and safety valves in routine monthly operating reports submitted to the NRC in accordance with the TS at the time. The industry proposed and the NRC accepted the elimination of the reporting requirements in TS for challenges to the pressurizer PORVs and safety valves in Revision 4 to TSTF-258, "Changes to Section 5.0, Administrative Controls" (Reference 6.8). The NRC staff’s acceptance of TSTF-258 and subsequent approval of plant-specific adoptions of TSTF-258 is based on the fact that the information on challenges to relief and safety valves is not used in the evaluation of the operating data, and that the information needed by the NRC is adequately addressed by the reporting requirements in 10 CFR 50.73, "Licensee event reports." Additionally, a low temperature over-pressure transient that violates the TS pressure - temperature limits would be immediately reportable under 10 CFR 50.72, “Immediate notification requirements for operating nuclear power reactors.”

Duke Energy proposes the deletion of this reporting requirement on the basis that the change is consistent with the NUREG-1431 STS and the report is a duplicative requirement. While the requirement to report mitigation of a pressure transient is not contained within NUREG-1431 STS 3.4.12, Low Temperature Overpressure Protection (LTOP) System, the requirements in 10 CFR 50.72 and 10 CFR 50.73 adequately address the reporting of low temperature overpressure transients. As such, elimination of this reporting requirement from the HNP TS is acceptable.

TS 3/4.5.2 and TS 3/4.5.3 – Emergency Core Cooling System

HNP TS 3/4.5.2 and TS 3/4.5.3 both require submitting a special report within 90 days if ECCS actuates and injects water into the RCS. Duke Energy proposes to delete these reporting requirements, consistent with NUREG-1431 STS 3.5.2, ECCS - Operating, and STS 3.5.3, ECCS - Shutdown, where neither require reporting an actuation of the ECCS. Moreover, the TS special reporting requirement is duplicative of the requirement in 10 CFR 50.73(a)(2)(iv) to report an actuation of the ECCS, except the TS report is provided a 90-day reporting time as compared to the timelier 10 CFR 50.73 requirement of within 60 days. Additionally, an immediate notification (within four hours) would be required through the event notification system (ENS) for an ECCS actuation that meets the criteria of 10 CFR 50.72(b)(2)(iv), an event that results or should have resulted in ECCS discharge into the RCS as a result of a valid signal.

While 10 CFR 50.73 requires an LER in the event of an ECCS actuation, it does not cover all the required content currently specified in HNP TS 3.5.2 and TS 3.5.3. Specifically, these TS require that the special reports include “total accumulated actuation cycles to date.” As discussed in HNP UFSAR section 3.9.1.1, Table 3.9.1-1 provides the limiting design transients and the number of cycles of each transient that is normally used for fatigue evaluations. HNP procedure OMM-013, “Cycle and Transient Monitoring Program,” establishes a tracking program for component cycles and performance of design transients to prevent exceeding design cyclic or transient limits. Corrective actions are initiated when the rate of accumulation of any transient or cycle is such that the cyclic or transient limit would be reached before the design lifetime of the plant.

Duke Energy proposes the deletion of the reporting requirements on the basis that the change is consistent with NUREG-1431 STS and that the ECCS actuations are adequately addressed per the requirements of 10 CFR 50.72 and 10 CFR 50.73, making the TS reporting requirements duplicative in nature. Further, the monitoring provided by HNP procedure OMM-013 provides reasonable assurance that ECCS actuations will be appropriately tracked, assuring that the integrity of the components will be maintained within design limits. As such, Duke Energy concludes that deletion of TS 3.5.2, ACTION b, and TS 3.5.3, ACTION c is acceptable.

TS 3/4.6.1.6 – Containment Vessel Structural Integrity

HNP TS SR 4.6.1.6.2 currently requires submitting a special report if the results of inspections of the containment interior and exterior surfaces do not meet acceptance standards. Duke Energy proposes to delete this requirement to submit a special report.

The structural integrity of the exposed accessible interior and exterior surfaces of the containment vessel, including the liner plate, is required to be determined during the visual inspection performed in accordance with the Containment Leakage Rate Testing Program, as described in HNP TS 6.8.4.k. This program implements the requirements of 10 CFR 50.54(o) and 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled

Power Reactors," which only requires that the results of the containment integrated leak rate testing, including the visual inspection, be documented and readily available on site for inspection by the NRC. Additional inspections shall be conducted in accordance with Subsections IWE and IWL of the ASME Boiler and Pressure Vessel Code, Section XI. The TS requirement to submit a special report is not necessary to meet the federal regulation or to ensure safe plant operation. This proposed change does not alter or change any existing reporting obligations required by 10 CFR 50 and maintains consistency with applicable regulatory requirements. As such, elimination of this reporting requirement from the HNP TS is acceptable.

TS 3/4.7.9 – Sealed Source Contamination

Per TS, HNP is required to submit a report on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microCurie of removable contamination. Duke Energy is proposing the deletion of this reporting requirement.

HNP TS 3/4.7.9 requires sealed sources to be free of greater than or equal to 0.005 microCurie of removable contamination, where the limitation on removable contamination is required by 10 CFR 31.5(c)(5). This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. With removable contamination above the limit, the TS required Action is to immediately withdraw the sealed source from use and either decontaminate and repair it, or dispose of it in accordance with NRC regulations.

Duke Energy is not proposing any changes to the TS 3/4.7.9 requirements other than deletion of the annual reporting requirement. As such, the current required actions related to the sealed sources are retained in the TS. Furthermore, identification of a source exceeding the allowable limits would be entered into the corrective action program, which is subject to audit by the NRC. A report submitted per TS 3/4.7.9 would not provide any new or additional information for the NRC staff from that captured in the corrective action program. Additionally, the report is not necessary to ensure safe plant operation, nor does it provide a safety benefit. As it relates to STS, NUREG-1431 does not contain a similar TS or reporting requirement related to sealed sources. Moreover, the requirement of SR 4.7.9.3 to submit the annual report is not necessary to comply with the 10 CFR 50.36(c)(3) regulation associated with surveillance requirements.

Duke Energy concludes that deletion of the reporting requirement is appropriate.

TS 5.3.1 – Fuel Assemblies

HNP TS 5.3.1 currently requires submitting a special report within 30 days of cycle startup if more than 30 individual rods in the core or ten fuel rods in any one fuel assembly are replaced in a refueling. Duke Energy is proposing the deletion of this reporting requirement, as consistent with NUREG-1431 STS, which does not include a similar reporting requirement.

The special report does not seek NRC approval and is not required to assure safe operation of the facility. Each cycle-specific core reload is implemented using NRC-approved methodologies and Duke Energy’s design change process. As such, the core design is implemented without prior NRC approval if determined acceptable under 10 CFR 50.59. Information regarding core design changes implemented during refueling is available to the NRC through resident inspector activities and documentation generated per the requirements of 10 CFR 50.59, which are subject to NRC review.

Duke Energy concludes that deletion of the reporting requirement is appropriate.

TS 6.9.1.1 – Startup Report

HNP TS 6.9.1.1 currently requires periodic submittal of a Startup Report to the NRC, providing a summary of plant startup and power ascension testing. Duke Energy is proposing the deletion of this reporting requirement.

The NRC published RG 1.16 in October of 1971 to provide an acceptable basis for meeting the reporting requirements listed in facility operating license Appendix A, "Technical Specifications Related to Health and Safety." Including the plant Startup Report, this RG provided a description of each of the routine reports that licensees are required to submit to demonstrate compliance with TS reporting requirements. As mentioned previously, however, the NRC withdrew RG 1.16 in August 2009 as it was no longer needed on the basis that TS reporting requirements are contained in 10 CFR 50 as well as other parts of 10 CFR Chapter 1.

Additionally, NUREG-1431 Chapter 5, Administrative Controls, which contains guidance on the content and frequency of required reports, does not include a requirement to submit a Startup Report. Information submitted in the Startup Report does not seek NRC approval for plant operation and is readily available to the NRC for inspection by the NRC Resident Inspectors.

Duke Energy concludes that the deletion of the requirement to provide a Startup Report is appropriate.

TS 6.9.1.2 – Annual Reports

HNP TS 6.9.1.2.a requires annual reporting of the results of specific activity analyses in which the primary coolant exceeded the limits of TS 3.4.8, Specific Activity. This reporting requirement is no longer common within TS for nuclear power plants and is not within NUREG-1431 STS. As this is the only annual report listed under TS 6.9.1.2, Duke Energy proposes the deletion of the annual report TS requirement in its entirety.

Duke Energy reports RCS specific activity by means of the performance indicators (PIs) under the Reactor Oversight Process (ROP). As part of the ROP PI Program, HNP currently provides monthly reactor coolant specific activity data on a quarterly basis to the NRC in accordance with Regulatory Issue Summary (RIS) 2000-08, Revision 1, "Voluntary Submission of Performance Indicator Data" [Reference 6.9] following the guidelines provided in NEI 99-02, "Regulatory Assessment Performance Indicator Guideline" [Reference 6.10]. As such, the RCS specific activity is provided more frequently than required by the annual report, regardless of if the TS limit is exceeded.

Unless the activity exceeds the TS 3.4.8 limit, the annual specific activity analysis report does not provide any information different from the performance indicator under the ROP. Deletion of the report is intended to eliminate unnecessary use of NRC and plant resources in providing redundant data. The limit is not expected to be exceeded unless a significant fuel issue occurs.

In the event that a fuel failure occurs, the condition would be captured in the corrective action program and evaluated for reportability and for the cause, with the information available for NRC review through its inspection program. Additionally, should the conditions warrant a report be submitted in accordance with 10 CFR 50.72 and 10 CFR 50.73, the appropriate information would be provided to the NRC in a timely manner as required per the regulation(s).

Providing an annual report on RCS activity for the previous year does not ensure continued safe operation of the facility, and the NUREG-1431 STS do not require such a report. Further, RCS activity is provided to the NRC on a quarterly basis under the ROP. As such, Duke Energy concludes that the annual reporting of specific activity analyses in accordance with HNP TS 6.9.1.2.a is no longer warranted.

Providing Additional Clarification

TS 3/4.4.4 – Relief Valves

Duke Energy is proposing to add clarification to HNP TS 3/4.4.4 to remove any potential for confusion related to which portions of ACTION b are appropriate for isolated PORVs under ACTION c for inoperable block valves. Currently, the HNP TS 3/4.4.4 ACTION c indicates that the licensee is to apply ACTION b, as appropriate, for the isolated PORV(s). The intent of this action is to apply the restoration times associated with ACTIONS b.1 and b.2, dependent on the number of inoperable PORVs. However, in the event of an inoperable block valve being stuck open, an improper application of ACTION b would result in an erroneous entrance into TS LCO 3.0.3 due to the licensee being unable to satisfy the action to restore the PORV within 1 hour or close the block valve. This would be an unnecessary entrance into TS LCO 3.0.3, which requires the licensee to initiate action within 1 hour to place the unit in a MODE in which the specification does not apply by placing it, as applicable, in at least HOT STANDBY within the next 6 hours, at least HOT SHUTDOWN within the following 6 hours, and at least COLD SHUTDOWN within the subsequent 24 hours.

Duke Energy is proposing additional clarification to align with the content provided in the Bases for NUREG-1431 STS 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)," ACTIONS C.1 and C.2, which states:

> "*If one block valve is inoperable, then it is necessary to either restore the block valve to OPERABLE status within the Completion Time of 1 hour or place the associated PORV in the closed position. The prime importance for the capability to close the block valve is to isolate a stuck open PORV. Therefore, if the block valve cannot be restored to OPERABLE status within 1 hour, the Required Action is to place the PORV in the closed position and remove power from the solenoid to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck open PORV at a time that the block valve is inoperable. <u>The actions for an inoperable PORV are not entered due to these actions, however, the associated PORV is inoperable and must be included in subsequent inoperability determinations.</u> The Completion Time of 1 hour is reasonable, based on the small potential for challenges to the system during this time period, and provides the operator time to correct the situation." [Emphasis added].*

In clarifying the application of ACTIONS b.1 or b.2, HNP would avoid the potential of unnecessarily entering into TS LCO 3.0.3 and would continue to be afforded the appropriate amount of time to restore the block valve(s) and associated isolated PORV(s) to OPERABLE status. In the event of not meeting the Completion Time, HNP would continue to be required to place the unit in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, which is similarly aligned with the NUREG-1431 STS requirements to be in MODE 3 within 6 hours and in MODE 4 within 12 hours.

Removal of Second Completion Time

TS 3/4.8.1.1 – A.C. Sources - Operating

As discussed in TSTF-439, Revision 2, the adoption of a second Completion Time was based on an NRC concern that a plant could continue to operate indefinitely with an LCO governing safety significant systems never being met by alternately meeting the requirements of separate Conditions. In 1991, there was no regulatory requirement or program which could prevent this misuse of the TS. However, that is no longer the case. With the promulgation of the Maintenance Rule, implementation of the Reactor Oversight Process, and the inclusion of administrative controls as discussed herein, there would exist strong disincentive to continued operation with concurrent multiple inoperabilities of the type the second Completion Times were designed to prevent.

<u>Maintenance Rule</u>

10 CFR 50.65(a)(1), "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" (i.e., the Maintenance Rule), requires each licensee to monitor the performance or condition of SSCs against licensee-established goals to ensure that the SSCs are capable of fulfilling their intended functions. If the performance or condition of an SSC does not meet established goals, appropriate corrective action is required to be taken. The NRC Resident Inspectors monitor the licensee's Corrective Action Program (CAP) and could take action if the licensee's maintenance program allowed the systems required by a single LCO to become concurrently inoperable multiple times. The performance and condition monitoring activities required by 10 CFR 50.65(a)(1) and (a)(2) would identify if poor maintenance practices resulted in multiple entries into the ACTIONS of the TS and unacceptable unavailability of these SSCs. The effectiveness of these performance monitoring activities, and associated corrective actions, is evaluated at least every refueling cycle, not to exceed 24 months per 10 CFR 50.65(a)(3).

Under the TS, the Completion Time for one system is not affected by other inoperable equipment. The second Completion Times were an attempt to influence the Completion Time for one system based on the condition of another system, if the two systems were required by the same LCO. However, 10 CFR 50.65(a)(4) is a much better mechanism to apply this influence as the Maintenance Rule considers all inoperable risk-significant equipment, not just the one or two systems governed by the same LCO.

Under 10 CFR 50.65(a)(4), the risk impact of all inoperable risk-significant equipment is assessed and managed when performing preventative or corrective maintenance. The risk assessments are conducted using the procedures and guidance endorsed by Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" (Reference 6.11). Regulatory Guide 1.160 endorses the Revision 4A of NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at

Nuclear Power Plants" (Reference 6.12). These documents address general guidance for conduct of the risk assessment, quantitative and qualitative guidelines for establishing risk management actions, and example risk management actions. These include actions to:

- plan and conduct other activities in a manner that controls overall risk,
- increased risk awareness by shift and management personnel,
- reduce the duration of the condition,
- minimize the magnitude of risk increases through the establishment of backup success paths or compensatory measures,
- and determination that the proposed maintenance is acceptable.

This comprehensive program provides much greater assurance of safe plant operation than the Second Completion Times in the TS.

Reactor Oversight Process

Nuclear Energy Institute (NEI) 99-02, "Regulatory Assessment Performance Indicator Guideline," describes the tracking and reporting of performance indicators to support the NRC's Reactor Oversight Process (ROP) (Reference 6.10). The NEI document is endorsed by Regulatory Issue Summary (RIS) 2001-11, "Voluntary Submission of Performance Indicator Data" (Reference 6.13). Section 2.2 of NEI 99-02 describes the Mitigating Systems Cornerstone and addresses Emergency AC Sources, which encompasses the AC Sources LCOs. Extended unavailability of these systems due to multiple entries into the ACTIONS would affect the NRC's evaluation of the licensee's performance under the ROP.

Administrative Controls

In addition to these programs, there is an existing action in the Duke Energy action tracking system for revision of OMM-014, “Operation of the Work Coordination Center,” to implement the guidance contained in TSTF-439, Revision 2. Specifically:

> It is possible to alternate between Technical Specification Conditions in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. However, doing so would be inconsistent with the basis of the Completion Times. Therefore, the maximum time allowed for any combination of Conditions that result in a single contiguous occurrence of failing to meet the LCO shall be limited. The Completion Times for those Conditions shall not be inappropriately extended.

OMM-014 provides instructions to track inoperable equipment and to ensure compliance with the LCO, including guidance for the identification, evaluation, tracking and initiation of reports required by LCOs and for use of the LCO Tracking Records as one method to document compliance with LCOs. This change will be part of the implementation process for this TS change.

Based on the above discussions, the concern regarding contiguous events of either an offsite or onsite A.C. source becoming inoperable and resulting in failure to meet the LCO is addressed by the system unavailability monitoring programs described above and the guidance being placed in HNP procedure OMM-014. As such, the TS can be simplified with the elimination of the second Completion Time with no detriment to plant safety.

4.0 Regulatory Evaluation

4.1 Applicable Regulatory Requirements/Criteria

The following NRC requirements and guidance documents are applicable to the proposed changes:

10 CFR 50 Appendix A, General Design Criterion 13

10 CFR Part 50 Appendix A, General Design Criterion (GDC) 13 states, “Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.”

HNP will continue to meet this criterion with the proposed changes. The monitors and interlocks in question will still be maintained within the specified surveillance testing frequency per SR 4.3.1.1, which is adequate to ensure continued reliable function of equipment. The SR being eliminated is an extraneous and unnecessary performance of the ACOT. The ability of the monitors to perform their required function will not be impacted.

10 CFR 50 Appendix A, General Design Criterion 20 through 25 and 29

GDC-20 requires that the protection system(s) shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

GDC-21 requires that the protection system(s) shall be designed for high functional reliability and testability.

GDC-22 through GDC-25 and GDC-29 require various design attributes for the protection system(s), including independence, safe failure modes, separation from control systems, requirements for reactivity control malfunctions, and protection against anticipated operational occurrences.

Compliance with these GDCs are not impacted by the proposed changes, including the removal of the extraneous surveillance tests. SRs as required by RTS instrumentation will remain and ensure continued compliance with these regulations. The proposed change only impacts testing frequency, and not testing methods.

10 CFR 50.36, “Technical specifications”

10 CFR 50.36(c)(2) states, “When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the

technical specifications until the condition can be met." The proposed changes continue to meet the requirements of this regulation.

10 CFR 50.36(c)(3) establishes requirements related to SR and states, "Surveillance requirements 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, and that the limiting conditions for operation will be met." The proposed changes continue to meet the requirements of this regulation.

10 CFR 50.36(c)(5) states, "Administrative controls are the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. Each licensee shall submit any reports to the Commission pursuant to approved technical specifications as specified in § 50.4." The proposed changes continue to meet the requirements of this regulation.

10 CFR 50.54, "Condition of licenses"

10 CFR 50.54 specifies requirements regarding responsibilities and staffing of license operators. Specifically, 10 CFR 50.54(o) establishes that primary reactor containments for water cooled power reactors shall be subject to the requirements set forth in appendix J to this part. The proposed changes continue to meet the requirements of this regulation.

10 CFR 50.55a, "Codes and standards"

10 CFR 50.55a requires that nuclear plants meet the requirements of the ASME Boiler and Pressure Vessel Code and the ASME Code for Operation and Maintenance of Nuclear Power Plants. Section b(2)(viii), "Section XI condition: Concrete containment examinations," addresses the application of Subsection IWL. The proposed changes continue to meet the requirements of this regulation.

10 CFR 50.55a(h) requires that protection systems meet IEEE 279-1971. Sections 4.9 - 4.11 of IEEE 279-1971 discuss testing provisions for protection systems. The proposed changes do not result in a change to the protection system instrumentation such that the above regulatory requirements or criteria would not be met. The proposed change to eliminate redundant testing only impacts testing frequency, and not testing methods.

10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants"

The regulations of 10 CFR 50.65 establish a performance-based rule to ensure that nuclear power plant structures, systems, and components (SSCs) will be maintained so that they will perform their intended function when required. The proposed changes do not impact the ability of HNP to continue to meet the requirements of this regulation.

10 CFR 50.72, "Immediate notification requirements for operating nuclear power reactors," and 10 CFR 50.73, "Licensee event report system"

10 CFR 50.72 and 10 CFR 50.73 provide requirements for making prompt notifications and submitting written reports to the NRC, respectively. These regulations cover events including

emergency system actuations, unanalyzed conditions, and inability of equipment to fulfill a safety function. The proposed changes do not impact the ability of HNP to continue to meet the requirements of this regulation.

Regulatory Guide 1.118, Revision 2

Regulatory Guide 1.118, Revision 2, discusses acceptable methods for testing protection systems. The proposed change to eliminate redundant testing only impacts testing frequency, and not testing methods.

The HNP TS amended as proposed will comply with all applicable regulatory requirements and guidance.

4.2 Precedent

Removal of Redundant Testing Requirements

The NRC previously approved a change to Salem Nuclear Generating Station TS via letter dated October 13, 2006 (Reference 6.14), that eliminated the “within 12 hours” restriction for performing channel functional tests on the power range and intermediate range nuclear monitors prior to initiating physics tests or low flow tests. These changes were based on TSTF-108 and NUREG-1431.

The NRC also approved a change to delete the surveillance requirements to perform a channel functional test of the source range neutron flux monitor within 8 hours prior to the initial start of core alterations. This change is outside the scope of this LAR for HNP.

Removal of Extraneous Reporting Requirements

The NRC previously approved a change to TS for Turkey Point Nuclear Generating Unit Nos. 3 and 4 via letter dated March 19, 2018 (Reference 6.15), that removed certain reporting requirements. Specifically, the amendment addressed the deletion of the following reporting requirements as applicable to this LAR:

- TS 3.1.1.3 – Moderator Temperature Coefficient
- TS 3.4.9.4 – Overpressure Mitigating Systems
- TS 3.5.2 – ECCS Subsystems – T_{AVG} Greater Than or Equal to 350°F
- TS 3.5.3 – ECCS Subsystems – T_{AVG} Less Than 350°F
- TS 3.6.1.6 – Containment Structural Integrity
- TS 3.7.7 – Sealed Source Contamination
- TS 5.3.1 – Fuel Assemblies
- TS 6.9.1.1 – Startup Report
- TS 6.9.1.2 – Annual Report

Similar to HNP, Turkey Point has not adopted the NUREG-1431 STS.

The NRC also approved changes to remove the completion time for restoring spent fuel pool water level, to address inoperability of one of the two parallel flow paths in the residual heat removal or safety injection headers for the Emergency Core Cooling Systems, and other administrative changes. These additional changes are outside the scope of this LAR for HNP.

Removal of Second Completion Time

The NRC previously approved a change to TS for St. Lucie Plant, Unit Nos. 1 and 2, via letter dated October 5, 2015 (Reference 6.16), for the deletion of the second completion time associated with their Containment Spray and Cooling Systems based on the adoption of TSTF-439, Revision 2. Similar to TS for St. Lucie, HNP TS are not based on the format and content of STS and do not contain a Use and Application Section. Whereas St. Lucie TS 3.6.2.1, “Containment Spray and Cooling Systems,” was their only TS that contained a second Completion Time, HNP TS 3.8.1.1 is the only TS for HNP that contains a second Completion Time. Accordingly, this is the only TS that is being revised in accordance with TSTF-439, Revision 2.

4.3 No Significant Hazards Consideration Determination Analysis

Pursuant to 10 CFR 50.90, Duke Energy Progress, LLC (Duke Energy), hereby requests a revision to the Technical Specifications (TS) for the Shearon Harris Nuclear Power Plant, Unit 1 (HNP). Specifically, the proposed change would revise TS 3/4.10.3, “Special Test Exceptions, Physics Tests,” and TS 3/4.10.4, “Special Test Exceptions, Reactor Coolant Loops,” to eliminate the “within 12 hours” restriction from Surveillance Requirement (SR) 4.10.3.2 for performing an Analog Channel Operational Test (ACOT) on the intermediate and power range neutron monitors prior to initiating physics tests and to eliminate the “within 12 hours” restriction from SR 4.10.4.2 for performing an ACOT on the intermediate range monitors, power range monitors, and P-7 interlock prior to initiating startup or physics tests, respectively. These changes are consistent with Technical Specification Task Force (TSTF) Traveler TSTF-108, Revision 1, “Eliminate the 12 hour COT [Channel Operational Test] on power range and intermediate range channels for Physics Test Exceptions” (Reference 6.2). The proposed change would also delete certain reporting requirements from TS, provide clarification to required actions of TS 3/4.4.4, “Relief Valves,” eliminate a second Completion Time associated with TS 3/4.8.1.1, “A.C. Sources – Operating,” and reflect the title changes of the Plant Nuclear Safety Committee (PNSC) to the On-Site Review Committee (ORC) and the Plant General Manager to the plant manager. The proposed change to eliminate a second Completion Time in TS 3/4.8.1.1 is aligned with the justification provided in NRC-approved TSTF-439, “Eliminate Second Completion Times Limiting Time From Discovery of Failure to Meet an LCO” (Reference 6.3). These changes are consistent with Revision 4 of NUREG-1431, “Standard Technical Specifications – Westinghouse Plants” (Reference 6.1).

Duke Energy has evaluated whether a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1) Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The actions, surveillance requirements, and administrative controls associated with the proposed changes to TS are not initiators of any previously-evaluated accidents. As such, the probability of these previously-evaluated accidents is not affected by the proposed changes.

Furthermore, the proposed changes do not affect the design, operational characteristics, function, or reliability of any plant structure, system or component (SSC). The capability of any operable TS-required SSC to perform its specified safety function is not impacted by the proposed changes. As such, the consequences of accidents previously evaluated in the Updated Final Safety Analysis Report are unaffected by the proposed changes because no change to any equipment response or accident mitigation scenario has resulted. The proposed changes will have no adverse effect on the availability, operability, or performance of the safety-related systems and components assumed to actuate in the event of a design basis accident or transient.

Additionally, the source term, containment isolation, and radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated are not affected by the proposed changes. The proposed changes do not increase the types or amounts of radioactive effluent that may be released offsite nor significantly increase individual or cumulative occupational/public radiation exposures. The proposed changes are consistent with the safety analysis assumptions and resultant consequences.

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

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

Response: No.

The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated in the Updated Final Safety Analysis Report. No new accident scenarios, failure mechanisms, or limiting single failures are introduced as a result of the proposed changes. Specifically, no new hardware is being added to the plant as part of the proposed changes, no existing equipment design or function is being modified, and no significant changes in operations are being introduced. No new equipment performance burdens are imposed.

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

3) Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The proposed changes do not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. They do not alter any safety analysis assumptions, initial conditions, or results of any accident analyses. The ability of any operable structure, system or component to perform its designated safety function is unaffected by the proposed changes. The proposed changes will not result in plant operation in a configuration outside the design basis. Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, Duke Energy concludes that the proposed changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusions

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

5.0 Environmental Consideration

Duke Energy has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined by 10 CFR 20, or it would change an inspection or surveillance requirement. However, the proposed changes do not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs be prepared in connection with the proposed amendment.

6.0 References

1. NUREG-1431, Revision 4, “Standard Technical Specifications, Westinghouse Plants, Volume 1, Specifications,” dated April 2012 (ADAMS Accession No. ML12100A222)
2. TSTF-108-A, Revision 1, “Eliminate the 12 hour COT on power range and intermediate range channels for Physics Test Exceptions,” dated May 2, 1997 *(*ADAMS Accession No. ML040480061)
3. TSTF-439, Revision 2, “Eliminate Second Completion Times Limiting Time From Discovery of Failure To Meet an LCO,” dated June 20, 2005 (ADAMS Accession No. ML051860296)
4. Letter from Ngoc B. Lee (NRC) to W. R. Robinson (CP&L), “Issuance of Amendment No. 51 to Facility Operating License No. NPF-63 Regarding TS 3/4 8.1, AC Sources, for Shearon Harris Nuclear Power Plant, Unit 1 (TAC No. M87819),” dated November 4, 1994 (ADAMS Accession No. ML020570308)
5. NRC letter to the TSTF, "Status of TSTF-439, 'Eliminate Second Completion Times Limiting Time from Discovery of Failure to Meet an LCO,'" January 11, 2006 (Adams Accession No. ML060120272)

6. NUREG-0452, Revision 4, “Standard Technical Specifications for Westinghouse Pressurized Water Reactors,” dated Fall 1981 (ADAMS Accession No. ML102590431)
7. NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: Light Water Reactor (LWR) Edition,” Section 16.0, “Technical Specifications,” Revision 3, March 2010
8. TSTF-258-A, Revision 4, “Changes to Section 5.0, Administrative Controls,” dated June 29, 1999 (ADAMS Accession No. ML040620102)
9. NRC Regulatory Issue Summary (RIS), RIS 2000-08, "Voluntary Submission of Performance Indicator Data", Revision 1, dated February 19, 2009 (ADAMS Accession No. ML083290153)
10. NEI Guideline, NEI 99-02, "Regulatory Assessment Performance Indicator Guideline," Revision 7, dated August 31, 2013 (ADAMS Accession No. ML13261A116)
11. U.S. Nuclear Regulatory Commission Regulatory Guide 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," dated May 2012 (ADAMS Accession No. ML113610098)
12. Nuclear Energy Institute (NEI), NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants", Revision 4A, April 26, 2011 (ADAMS Accession No. ML11116A198)
13. NRC Regulatory Issue Summary (RIS), RIS 2001-11, "Voluntary Submission of Performance Indicator Data", dated May 11, 2001 (ADAMS Accession No. ML011240144)
14. Letter from Stewart N. Bailey (NRC) to William Levis (PSEG), “Salem Nuclear Generating Station, Unit Nos. 1 and 2, Issuance of Amendments Re: Refueling Operations and Special Test Exceptions (TAC Nos. MC9337 and MC9338),” dated October 13, 2006 (ADAMS Accession Nos. ML062690462, ML062690480, ML062900271)
15. Letter from Michael J. Wentzel (NRC) to Mano Nazar (Florida Power & Light Company), “Turkey Point Nuclear Generating Unit Nos. 3 and 4 – Issuance of Amendments Regarding the Elimination of Certain Technical Specifications Reporting Requirements (CAC Nos. MF9601 and MF9602; EPID L-2017-LLA-0213),” dated March 19, 2018 (ADAMS Accession No. ML18019A078)
16. Letter from Farideh E. Saba (NRC) to Mano Nazar (Nextra Energy), “St. Lucie Plant, Unit Nos. 1 and 2 – Issuance of Amendments Regarding Adoption of Technical Specification Task Force Traveler-439, Revision 2, ‘Eliminate Second Completion Times Limiting Time From Discovery of Failure to Meet an LCO’ (TAC Nos. MF5370 and MF5371),” dated October 5, 2015 (ADAMS Accession No. ML15251A094)

SERIAL RA-19-0123

ENCLOSURE 2

PROPOSED TECHNICAL SPECIFICATION CHANGES

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

RENEWED LICENSE NUMBER NPF-63

22 PAGES PLUS THE COVER

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be maintained within the limits specified in the CORE OPERATING LIMITS REPORT (COLR). The maximum positive limit shall be less than or equal to +5 pcm/°F for power levels up to 70% RATED THERMAL POWER and a linear ramp from that point to 0 pcm/°F at 100% RATED THERMAL POWER.

APPLICABILITY: Positive MTC Limit – MODES 1 and 2* only**.

Negative MTC Limit – MODES 1, 2, and 3 only**.

ACTION:

a. With the MTC more positive than the Positive MTC Limit specified in the COLR, operation in MODES 1 and 2 may proceed provided:

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to within the Positive MTC Limit specified in the COLR within 24 hours, or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

2. The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

~~3. A Special Report is prepared and submitted to the Commission, pursuant to Specification 6.9.2, within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the positive MTC to within its limit for the all rods withdrawn condition.~~

b. With the MTC more negative than the Negative MTC Limit specified in the COLR, be in HOT SHUTDOWN within 12 hours.

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*With k_{eff} greater than or equal to 1.

**See Special Test Exceptions Specification 3.10.3.

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

ACTION:

a. With one or more PORV(s) inoperable, because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one or more PORV(s) inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s), and

   1. With only one safety grade PORV OPERABLE, restore at least a total of two safety grade PORVs to OPERABLE status within the following 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

   2. With no safety grade PORVs OPERABLE, restore at least one safety grade PORV to OPERABLE status within 1 hour and follow ACTION b.1, above, with the time requirement of that ACTION statement based on the time of initial loss of the remaining inoperable safety grade PORV or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With one or more block valve(s) inoperable, within 1 hour: (1) restore the block valve(s) to OPERABLE status, or close the block valve(s) and remove power from the block valve(s), or close the PORV and remove power from its associated solenoid valve; and (2) apply ~~the~~ ACTION ~~b~~., above, as appropriate, for the isolated PORV(s).

d. The provisions of Specification 3.0.4 are not applicable.

b.1 or b.2

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.4 At least one of the following Overpressure Protection Systems shall be OPERABLE:

a. The Reactor Coolant System (RCS) depressurized with an RCS vent of greater than or equal to 2.9 square inches, or

* b. Two power-operated relief valves (PORVs) with setpoints which do not exceed the limits established in Figure 3.4-4.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to 325°F, MODE 5 and MODE 6 with the reactor vessel head on.

ACTION:

a. With one PORV inoperable in MODE 4, restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2.9 square inch vent within the next 8 hours.

b. With one PORV inoperable in MODES 5 or 6, either (1) restore the inoperable PORV to OPERABLE status within 24 hours, or (2) complete depressurization and venting of the RCS through at least a 2.9 square inch vent within the next 8 hours.

c. With both PORVs inoperable, depressurize and vent the RCS through at least a 2.9 square inch vent within 8 hours.

~~d. In the event either the PORVs or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.~~

d.

~~e~~. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.4.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the PORV actuation channel, but excluding valve operation, within 31 days prior to

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* Credit may only be taken for the setpoints when the RCS cold leg temperature ≥ 90°F.

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE Charging/safety injection pump,

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a Safety Injection signal and, upon being manually aligned, transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours* or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

~~b.~~ ~~In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.~~

-- NOTE --

*The ‘A’ Train ECCS subsystem is allowed to be inoperable for a total of 14 days only to allow for the implementation of design improvements on the ‘A’ Train ESW pump. The 14 days will be taken one time no later than October 29, 2016. During the period in which the ‘A’ Train ESW pump supply from the Auxiliary Reservoir or Main Reservoir is not available, Normal Service Water will remain available and in service to supply the ‘A’ Train ESW equipment loads until the system is ready for post maintenance testing. Allowance of the extended Completion Time is contingent on meeting the Compensatory Measures and Conditions described in HNP LAR submittal correspondence letter HNP-16-056.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

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

1. Verifying that the following valves are in the indicated positions with the control power disconnect switch in the "OFF" position, and the valve control switch in the "PULL TO LOCK" position:

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} LESS THAN 350°F

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE charging/safety injection pump,*

b. One OPERABLE RHR heat exchanger,

c. One OPERABLE RHR pump, and

d. An OPERABLE flow path capable of taking suction from the refueling water storage tank upon being manually realigned and transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the charging/safety injection pump or the flow path from the refueling water storage tank, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 24 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the residual heat removal heat exchanger or RHR pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods.

~~c.~~ ~~In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected Safety Injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.~~

* A maximum of one charging/safety injection pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 325°F.

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment vessel shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.6.1.

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

ACTION:

With the structural integrity of the containment vessel not conforming to the above requirements, restore the structural integrity to within the limits within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 Containment Vessel Surfaces. The structural integrity of the exposed accessible interior and exterior surfaces of the containment vessel, including the liner plate, shall be determined, during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2), by a visual inspection of these surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation. Additional inspections shall be conducted in accordance with Subsections IWE and IWL of the ASME Boiler and Pressure Vessel Code, Section XI.

~~4.6.1.6.2 Reports. Any abnormal degradation of the containment vessel structure detected during the above required inspections shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2 within 15 days. This report shall include a description of the condition of the concrete, the inspection procedure, the tolerances on cracking, and the corrective actions taken.~~

For information only.
No changes to this page.

3/4.7.9 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.9 Each sealed source (excluding startup sources and fission detectors previously subjected to core flux) containing radioactive material either in excess of 100 microCuries of beta and/or gamma emitting material or 10 microCuries of alpha emitting material shall be free of greater than or equal to 0.005 microCurie of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. With a sealed source having removable contamination in excess of the above limits, immediately withdraw the sealed source from use and either:

    1. Decontaminate and repair the sealed source, or

    2. Dispose of the sealed source in accordance with Commission Regulations.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.9.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microCurie per test sample.

4.7.9.2 Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequency described below.

a. Sources in use - At least once per 6 months for all sealed sources containing radioactive materials:

    1. With a half-life greater than 30 days (excluding Hydrogen 3), and

    2. In any form other than gas.

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use; and

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

~~4.7.9.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microCurie of removable contamination.~~

For information only.
No changes to this page.

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two separate and independent diesel generators, each with:

1. A separate day tank containing a minimum of 1457 gallons of fuel,

2. A separate main fuel oil storage tank containing a minimum of 100,000 gallons of fuel, and

3. A separate fuel oil transfer pump.

c. Automatic Load Sequencers for Train A and Train B.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one offsite circuit of 3.8.1.1.a inoperable:

1. Perform Surveillance Requirement 4.8.1.1.1.a within 1 hour and once per 8 hours thereafter; and

2. Restore the offsite circuit to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

3. Verify required feature(s) powered from the OPERABLE offsite A.C. source are OPERABLE. If required feature(s) powered from the OPERABLE offsite circuit are discovered to be inoperable at any time while in this condition, restore the required feature(s) to OPERABLE status within 24 hours from discovery of inoperable required feature(s) or declare the redundant required feature(s) powered from the inoperable A.C. source as inoperable.

b. With one diesel generator of 3.8.1.1.b inoperable:

1. Perform Surveillance Requirement 4.8.1.1.1.a within 1 hour and once per 8 hours thereafter; and

*2. Within 24 hours, determine the OPERABLE diesel generator is not inoperable due to a common cause failure or perform Surveillance Requirement 4.8.1.1.2.a.4#; and

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* This ACTION is required to be completed regardless of when the inoperable EDG is restored to OPERABILITY.

Activities that normally support testing pursuant to 4.8.1.1.2.a.4, which would render the diesel inoperable (e.g., air roll), shall not be performed for testing required by this ACTION statement.

For information only.
No changes to this page.

LIMITING CONDITION FOR OPERATION

ACTION (Continued):

3. Restore the diesel generator to OPERABLE status within 72 hours** or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

4. Verify required feature(s) powered from the OPERABLE diesel generator are OPERABLE. If required feature(s) powered from the OPERABLE diesel generator are discovered to be inoperable at any time while in this condition, restore the required feature(s) to OPERABLE status within 4 hours from discovery of inoperable required feature(s) or declare the redundant required feature(s) powered from the inoperable A.C. source as inoperable.

c. With one offsite circuit and one diesel generator of 3.8.1.1 inoperable:

NOTE: Enter applicable Condition(s) and Required Action(s) of LCO 3/4.8.3, ONSITE POWER DISTRIBUTION - OPERATING, when this condition is entered with no A.C. power to one train.

1. Restore one of the inoperable A.C. sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

2. Following restoration of one A.C. source (offsite circuit or diesel generator), restore the remaining inoperable A.C. source to OPERABLE status pursuant to requirements of either ACTION a or b, based on the time of initial loss of the remaining A.C. source.

**The ‘A’ diesel generator is allowed to be inoperable for a total of 14 days only to allow for the implementation of design improvements on the ‘A’ Train ESW pump. The 14 days will be taken one time no later than October 29, 2016. During the period in which the ‘A’ Train ESW pump from the Auxiliary Reservoir or Main Reservoir is not available, Normal Service Water will remain available and in service to supply the ‘A’ Train ESW equipment until the system is ready for post maintenance testing. Allowance of the extended Completion Time is contingent on meeting the Compensatory Measures and Conditions described in HNP LAR submittal correspondence HNP-16-056.

ACTION (Continued):

d. With two of the required offsite A.C. sources inoperable:

   1. Restore one offsite circuit to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

   2. Verify required feature(s) are OPERABLE. If required feature(s) are discovered to be inoperable at any time while in this condition, restore the required feature(s) to OPERABLE status within 12 hours from discovery of inoperable required feature(s) or declare the redundant required feature(s) inoperable.

   3. Following restoration of one offsite A.C. source, restore the remaining offsite A.C. source in accordance with the provisions of ACTION a with the time requirement of that ACTION based on the time of initial loss of the remaining inoperable A.C. source.

e. With two of the required diesel generators inoperable:

   1. Perform Surveillance Requirement 4.8.1.1.1.a within 1 hour and once per 8 hours thereafter; and

   #2. Restore one of the diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

   3. Following restoration of one diesel generator, restore the remaining diesel generator in accordance with the provisions of ACTION b with the time requirement of that ACTION based on the time of initial loss of the remaining inoperable diesel generator.

f. With three or more of the required A.C. sources inoperable:

   1. Immediately enter Technical Specification 3.0.3.

   2. Following restoration of one or more A.C. sources, restore the remaining inoperable A.C. sources in accordance with the provisions of ACTION a,b,c,d and/or e as applicable with the time requirement of that ACTION based on the time of initial loss of the remaining inoperable A.C. sources.

ADD: Deleted.

g. ~~With contiguous events of either an offsite or onsite A.C. source becoming inoperable and resulting in failure to meet the LCO:~~

   ~~1. Within 6 days, restore all A.C. sources required by 3.8.1.1 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

#Activities that normally support testing pursuant to 4.8.1.1.2.a.4, which would render the diesel inoperable (e.g., air roll), shall not be performed for testing required by this ACTION statement.

3/4.10.3 PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.3, 3.1.1.4, 3.1.3.1, 3.1.3.5, and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set at less than or equal to 25% of RATED THERMAL POWER, and

c. The Reactor Coolant System lowest operating loop temperature (T_{avg}) is greater than or equal to 541°F.

APPLICABILITY: MODE 2.

ACTION:

a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the Reactor trip breakers.

b. With a Reactor Coolant System operating loop temperature (T_{avg}) less than 541°F, restore T_{avg} to within its limit within 15 minutes or be in at least HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at the frequency specified in the Surveillance Frequency Control Program during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range channel shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST ~~within 12 hours~~ prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 541°F at the frequency specified in the Surveillance Frequency Control Program during PHYSICS TESTS.

3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.4.1.1 may be suspended during the performance of startup and PHYSICS TESTS in MODE 1 or 2 provided:

a. The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set less than or equal to 25% of RATED THERMAL POWER.

APPLICABILITY: During operation below the P-7 Interlock Setpoint.

ACTION:

With the THERMAL POWER greater than the P-7 Interlock Setpoint, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 Interlock Setpoint at the frequency specified in the Surveillance Frequency Control Program during startup and PHYSICS TESTS.

4.10.4.2 Each Intermediate and Power Range channel, and P-7 Interlock shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST ~~within 12 hours~~ prior to initiating startup and PHYSICS TESTS.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The containment building is designed and shall be maintained for a maximum internal pressure of 45.0 psig and a peak air temperature of 380°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 157 fuel assemblies with each fuel assembly normally containing 264 fuel rods clad with Zircaloy-4 or M5. Limited substitution of fuel rods by filler rods (consisting of Zircaloy-4 or M5 clad stainless steel or zirconium), or vacancies may be made in fuel assemblies if justified by a cycle specific evaluation. ~~Should more than a total of 30 fuel rods or more than 10 fuel rods in any one assembly be replaced per refueling, a Special Report describing the number of rods replaced will be submitted to the Commission, pursuant to Specification 6.9.2, within 30 days after cycle startup.~~ Each fuel rod shall have a nominal active fuel length of 144 inches. The initial core loading shall have a maximum enrichment of 3.5 weight percent U 235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 5.0 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 52 shutdown and control rod assemblies. The shutdown and control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80% silver, 15% indium, and 5% cadmium, or 95% hafnium with the remainder zirconium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The Reactor Coolant System is designed and shall be maintained:

a. In accordance with the Code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the Reactor Coolant System is approximately 10,300 cubic feet at a nominal T_{avg} of 588.8°F.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

plant manager

6.1.1 The ~~Plant General Manager~~ shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Superintendent-Shift Operations (or, during his absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Vice President-Harris Nuclear Plant shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

6.2.1 Onsite and Offsite Organization

An onsite and an offsite organization shall be established for unit operation and corporate management. The onsite and offsite organization shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility and communication shall be established and defined from the highest management levels through intermediate levels to and including all operating organization positions. Those relationships shall be documented and updated, as appropriate, in the form of organizational charts. These organizational charts will be documented in the FSAR and updated in accordance with 10 CFR 50.71(e).

b. There shall be an individual executive position (corporate officer) in the offsite organization having corporate responsibility for overall plant nuclear safety. This individual shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support in the plant so that continued nuclear safety is assured.

c. There shall be an individual management position in the onsite organization having responsibility for overall unit safe operation and shall have control over those onsite resources necessary for safe operation and maintenance of the plant.

d. Although the individuals who train the operating staff and those who carry out the quality assurance functions may report to the appropriate manager onsite, they shall have sufficient organizational freedom to be independent from operating pressures.

e. Although health physics individuals may report to any appropriate manager onsite, for matters relating to radiological health and safety of employees and the public, the healthy physics manager shall have direct access to that onsite individual having responsibility for overall unit management. Health physics personnel shall have the authority to cease any work activity when worker safety is jeopardized or in the event of unnecessary personnel radiation exposures.

6.6 REPORTABLE EVENT ACTION

ORC

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the ~~PNSC~~, and the results of this review shall be submitted to the Manager - Nuclear Assessment Section and the Vice President - Harris Nuclear Plant.

6.7 SAFETY LIMIT VIOLATION

6.7.1 Deleted.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Generic Letter No. 82-33;

c. Security Plan implementation;

d. Emergency Plan implementation;

e. PROCESS CONTROL PROGRAM implementation;

f. OFFSITE DOSE CALCULATION MANUAL implementation;

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the NRC in accordance with 10CFR50.4.

~~STARTUP REPORT~~

ADD: Deleted.

6.9.1.1 ~~A summary report of plant startup and power escalation testing shall be submitted following: (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.~~

~~The Startup Report shall address each of the tests identified in the Final Safety Analysis Report and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.~~

~~Startup Reports shall be submitted within: (1) 90 days following completion of the Startup Test Program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of Startup Test Program, and resumption or commencement of commercial operation), supplementary reports shall be submitted at least every 3 months until all three events have been completed.~~

~~ANNUAL REPORTS~~

ADD: Deleted.

6.9.1.2 ~~Annual Reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.~~

~~Reports required on an annual basis shall include:~~

~~ANNUAL REPORTS (Continued)~~

~~a.~~ ~~The results of specific activity analyses in which the reactor coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) reactor power history starting 48 hours prior to the first sample in which the limit was exceeded (in graphic and tabular format); (2) results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) cleanup flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) graph of the I-131 concentration (μCi/gm) and one other radio-iodine isotope concentration (μCi/gm) as a function of time for the duration of the specific activity above the steady-state level; and (5) the time duration when the specific activity of the reactor coolant exceeded the radioiodine limit.~~

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

6.9.1.3 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

6.9.1.4 The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous year shall be submitted prior to May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

HIGH RADIATION AREA (Continued)

6.13 PROCESS CONTROL PROGRAM (PCP)

Changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by FSAR Section 17.3. This documentation shall contain:

   1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

   2) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

ORC

b. Shall become effective after review and acceptance by the ~~PNSC~~ and the approval of the ~~Plant General Manager~~.

plant manager

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained as required by FSAR Section 17.3. This documentation shall contain:

   1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

OFFSITE DOSE CALCULATION MANUAL (Continued)

ORC

2) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b. Shall become effective after review and acceptance by the ~~PNSC~~ and the approval of the ~~Plant General Manager~~. plant manager

c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

6.15 Specification 6.15 has been deleted from Technical Specifications and has been relocated to the ODCM and PCP, as appropriate.

Page 6-29 has been deleted.

SERIAL RA-19-0123

ENCLOSURE 3

TECHNICAL SPECIFICATION BASES CHANGES

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

RENEWED LICENSE NUMBER NPF-63

4 PAGES PLUS THE COVER

3/4.8.1, 3/4.8.2, AND 3/4.8.3 A.C. SOURCES, D.C. SOURCES, AND ONSITE POWER DISTRIBUTION

The OPERABILITY of the A.C. and D.C power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety-related equipment required for: (1) the safe shutdown of the facility, and (2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix A to 10 CFR Part 50.

The switchyard is designed using a breaker-and-a-half scheme. The switchyard currently has seven connections with the Duke Energy transmission network; each of these transmission lines is physically independent. The switchyard has one connection with each of the two Startup Auxiliary Transformers and each SAT can be fed directly from an associated offsite transmission line. The Startup Auxiliary Transformers are the preferred power source for the Class 1E ESF buses. The minimum alignment of offsite power sources will be maintained such that at least two physically independent offsite circuits are available. The two physically independent circuits may consist of any two of the incoming transmission lines to the SATs (either through the switchyard or directly) and into the Class 1E system. As long as there are at least two transmission lines in service and two circuits through the SATs to the Class 1E buses, the LCO is met.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss-of-offsite power and single failure of the other onsite A.C. source. The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources," December 1974. There are additional ACTION requirements to verify that all required feature(s) that depend on the remaining OPERABLE A.C. sources as a source of emergency power, are also OPERABLE. These requirements allow a period of time to restore any required feature discovered to be inoperable, e.g. out-of-service for maintenance, to an OPERABLE status. If the required feature(s) cannot be restored to an OPERABLE status, the ACTION statement requires the redundant required feature, i.e. feature receiving power from an inoperable A.C. source, to be declared inoperable. The allowed operating times to restore an inoperable required feature to an OPERABLE status is based on the requirements in NUREG 1431. The term "verify", as used in these ACTION statements means to administratively check by examining logs or other information to determine the OPERABILITY of required feature(s). It does not mean to perform the Surveillance Requirement needed to demonstrate the OPERABILITY of the required feature(s).

ADD:
<INSERT>

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that: (1) the facility can be maintained in the shutdown or refueling condition for extended time periods, and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are based upon the recommendations of Regulatory Guides 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies," December 1979; 1.108, "Periodic Testing of Diesel

Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977 as modified in accordance with the guidance of IE Notice 85-32, April 22, 1985; and 1.137, "Fuel-Oil Systems for Standby Diesel Generators," Revision 1, October 1979. Proper shedding and sequencing of loads are required functions for Emergency Diesel Generator OPERABILITY. Pressure testing of the diesel generator fuel oil piping at 110% of the system design pressure will only be required on the isolable portions of (1) fuel oil transfer pump discharge piping to the day tank, (2) fuel oil supply from the day tank to the diesel vendor-supplied piping, and (3) fuel oil return piping from the diesel vendor-supplied piping to the day tank regulator valve. The exemptions allowed by ASME Code Section XI will be invoked for the atmospheric day tanks and non-isolable piping. The surveillance frequencies are controlled in the Surveillance Frequency Control Program.

The inclusion of the loss of generator potential transformer circuit lockout trip is a design feature based upon coincident logic and is an anticipatory trip prior to diesel generator overspeed. In TS 4.8.1.1.2.f.13, the phrase “all diesel generator trips” refers to automatic protective trips.

The Surveillance Requirements for demonstrating the OPERABILITY of the station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations." The performance test supporting the Surveillance Requirement incorporates the guidance of IEEE Std 450-2010. The surveillance frequencies are controlled in the Surveillance Frequency Control Program.

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values, and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates, and compares the battery capacity at that time with the rated capacity.

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage, and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and 0.015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than 0.020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than 0.010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-2 is permitted for up to 7 days. During this 7-day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than 0.020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than 0.040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

LCOs 3.8.3.1 and 3.8.3.2 include requirements for energizing 118 VAC vital buses from the associated inverters connected to 125 VDC buses. In the event the 118 VAC vital buses are not energized by the inverters connected to the 125 VDC buses, system design provides for energizing the 118 VAC buses from the Bypass Source or the Alternate Power Supply. The Bypass Source is regulated, transfer to the source is automatic within the inverters, and operation on the Bypass Source requires entry into LCO 3.8.3.1 Action ‘c’ or LCO 3.8.3.2 Action, depending on the OPERATIONAL MODE. The Alternate Power Supply is unregulated and the voltage may not be sufficient to support loads as documented in calculation E-6007. Operation on the Alternate Power Supply, requires entry into LCO 3.8.3.1 Action ‘b’ or LCO 3.8.3.2 Action, depending on the OPERATIONAL MODE.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Containment electrical penetrations and penetration conductors are protected by either deenergizing circuits not required during reactor operation or by demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers during periodic surveillance.

The Surveillance Requirements applicable to lower voltage circuit breakers provide assurance of breaker reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker. Each manufacturer's molded case and metal case circuit breakers are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers are tested. If a wide variety exists within any manufacturer's brand of circuit breakers, it is necessary to divide that manufacturer's breakers into groups and treat each group as a separate type of breaker for surveillance purposes. For surveillance 4.8.4.1.a.1.c and 4.8.4.1.a.2, testing of the breakers includes a representative sample of 10% of each type of breaker as described in the table below.

Types
15-Amp(A)
30A-40A
50A
60A
70A-90A
100-110A
125-150A
225A

The bypassing of the motor-operated valves thermal overload protection during accident conditions by integral bypass devices ensures that safety-related valves will not be prevented from performing their function. The Surveillance Requirements for demonstrating the bypassing of the thermal overload protection during accident conditions are in accordance with Regulatory Guide 1.106, “Thermal Overload Protection for Electric Motors on Motor Operated Valves,” Revision 1, March 1977.

<INSERT>

It is possible to alternate between Technical Specification Conditions in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. However, doing so would be inconsistent with the basis for Completion Times. Therefore, the maximum time allowed for any combination of Conditions that result in a single contiguous occurrence of failing to meet the LCO shall be limited. The Completion Times for those Conditions shall not be inappropriately extended.