



Kim Maza
Site Vice President
Harris Nuclear Plant
5413 Shearon Harris Rd
New Hill, NC 27562-9300

984-229-2512

10 CFR 50.90

February 24, 2021
Serial: RA-20-0252

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 Remove Extraneous Content and Requirements from the Operating License and Technical Specifications

Ladies and Gentlemen:

In accordance with the provisions of 10 CFR 50.90, Duke Energy Progress, LLC (Duke Energy), hereby requests a revision to the Shearon Harris Nuclear Power Plant, Unit 1 (HNP), Operating License and Technical Specifications (TS). The proposed amendment would remove License Condition 2.G, "Reporting to the Commission," which requires Duke Energy to report any violations of Operating License Section 2.C within twenty-four hours to the Nuclear Regulatory Commission Operations Center via the Emergency Notification System with a written follow-up within 30 days. Additionally, the proposed change would delete HNP TS 3/4.4.10, "Structural Integrity," revise Administrative Control TS 6.1.2 to eliminate the annual management directive requirement, and revise TS Table 4.3-2, "Engineered Safety Features Actuation System Instrumentation Surveillance Requirements," to remove an overly restrictive requirement that impedes the full application of the Surveillance Frequency Control Program for a specific subset of relays. The proposed amendment would also revise HNP TS to remove the Index and place it under licensee control.

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.

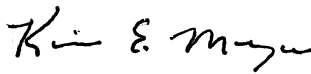
This letter contains no regulatory commitments.

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 February 24, 2021.

Sincerely,



Kim Maza
Site Vice President
Harris Nuclear Plant

Enclosures:

1. Evaluation of the Proposed Changes
2. Proposed Technical Specification Changes (Mark-up)
3. Proposed Technical Specification Bases Changes (Mark-up)

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

U.S. Nuclear Regulatory Commission
Serial: RA-20-0252
Enclosure 1

ENCLOSURE 1

EVALUATION OF THE PROPOSED CHANGES

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

RENEWED LICENSE NUMBER NPF-63

18 PAGES PLUS THE COVER

Evaluation of the Proposed Changes

License Amendment Request to Remove Extraneous Content and Requirements from Operating License and Technical Specifications

1.0 SUMMARY DESCRIPTION

In accordance with the provisions of 10 CFR 50.90, Duke Energy Progress, LLC (Duke Energy), hereby requests a revision to the Shearon Harris Nuclear Power Plant, Unit 1 (HNP), Operating License and Technical Specifications (TS). The proposed amendment would remove License Condition 2.G, "Reporting to the Commission," which requires Duke Energy to report any violations of Operating License Section 2.C within twenty-four hours to the Nuclear Regulatory Commission (NRC) Operations Center via the Emergency Notification System with a written follow-up within 30 days. Additionally, the proposed change would delete HNP TS 3/4.4.10, "Structural Integrity," revise Administrative Control TS 6.1.2 to eliminate the annual management directive requirement, and revise TS Table 4.3-2, "Engineered Safety Features Actuation System Instrumentation Surveillance Requirements," to remove an overly restrictive requirement that impedes the full application of the Surveillance Frequency Control Program for a specific subset of relays. The proposed amendment would also revise HNP TS to remove the Index and place it under licensee control.

2.0 DETAILED DESCRIPTION

2.1 Current Operating License and Technical Specifications

2.1.1 Operating License Section 2.G

The current requirements of the license condition are as follows:

G. Reporting to the Commission

Except as otherwise provided in the Technical Specifications or Environmental Protection Plan, Duke Energy Progress, LLC shall report any violations of the requirements contained in Section 2.C of this license in the following manner: initial notification shall be made within twenty-four (24) hours to the NRC Operations Center via the Emergency Notification System with written follow-up within 30 days in accordance with the procedures described in 10 CFR 50.73 (b), (c) and (e).

The existing conditions in Section 2.C that are subject to the current reporting requirement consist of the following:

(1) Maximum Power Level

Duke Energy Progress, LLC is authorized to operate the facility at reactor core power levels not in excess of 2948 megawatts thermal (100 percent rated core power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 179, are hereby incorporated into this license. Duke Energy Progress, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

Duke Energy Progress, LLC shall comply with the antitrust conditions delineated in Appendix C to this license.

Conditions (4) through (7) are associated with reports following the first refueling outage and are not subject to the reporting requirements of Section 2.G. Additionally, Condition (8) was previously deleted.

(9) Formal Federal Emergency Management Agency Finding

In the event that the NRC finds that the lack of progress in completion of the procedures in the Federal Emergency Management Agency's final rule, 44 CFR Part 350, is an indication that a major substantive problem exists in achieving or maintaining an adequate state of emergency preparedness, the provisions of 10 CFR Section 50.54(s)(2) will apply.

(10) Fresh Fuel Storage

The following criteria apply to the storage and handling of new fuel assemblies in the Fuel Handling Building:

- (a) The minimum edge-to-edge distance between a new fuel assembly outside its shipping container or storage rack and all other new fuel assemblies shall be at least 12 inches.
- (b) New fuel assemblies shall be stored in such a manner that water would drain freely from the assemblies in the event of flooding and subsequent draining of the fuel storage area.

(11) Mitigation Strategy License Condition

Develop and maintain strategies for addressing large fires and explosions and that include the following key areas:

- (a) Fire fighting response strategy with the following elements:
 - 1. Pre-defined coordinated fire response strategy and guidance
 - 2. Assessment of mutual aid fire fighting assets
 - 3. Designated staging areas for equipment and materials
 - 4. Command and control
 - 5. Training of response personnel

- (b) Operations to mitigate fuel damage considering the following:
 - 1. Protection and use of personnel assets
 - 2. Communications
 - 3. Minimizing fire spread
 - 4. Procedures for implementing integrated fire response strategy
 - 5. Identification of readily-available pre-staged equipment
 - 6. Training on integrated fire response strategy
 - 7. Spent fuel pool mitigation measures

- (c) Actions to minimize release to include consideration of:
 - 1. Water spray scrubbing
 - 2. Dose to onsite responders

(12) Control Room Habitability

Upon implementation of Amendment No. 128 adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by Surveillance Requirement (SR) 4.7.6.g, in accordance with TS 6.8.4.o.3(i), the assessment of CRE habitability as required by TS 6.8.4.o.3(ii) and the measurement of CRE pressure as required by TS 6.8.4.o.4, shall be considered met. Following implementation:

- a) The first performance of SR 4.7.6.g, in accordance with Specification 6.8.4.o.3(i), shall be within the specified Frequency of 6 years, plus the 18-month allowance of SR 4.0.2, as measured from March 5, 2004, the date of the most recent successful tracer gas test, or within the next 18 months if the time period since the most recent successful tracer gas test is greater than 6 years.

- b) The first performance of the periodic assessment of CRE habitability, Specification 6.8.4.o.3(ii), shall be within 3 years, plus the 9-month allowance of SR 4.0.2, as measured from March 5, 2004, the date of the most recent successful tracer gas test, or within the next 9 months if the time period since the most recent successful tracer gas test is greater than 3 years.

- c) The first performance of the periodic measurement of CRE pressure, Specification 6.8.4.o.4, shall be within 18 months plus 138 days allowed by SR 4.0.2 as measured from October 13, 2006, the date of the most recent successful pressure measurement test.

2.1.2 Technical Specifications

The HNP TS are based upon the format and content of the NUREG-0452, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors," series. As a result, the HNP TS numbers and associated Bases numbers differ from those contained in NUREG-1431, "Standard Technical Specifications – Westinghouse Plants" (Revision 4, ADAMS Accession No. ML12100A222). Additionally, the HNP TS maintains an Index of the Technical Specifications and their respective Bases.

TS 3/4.4.10 – Structural Integrity

The purpose of TS 3/4.4.10 is to specify the requirements for maintaining the structural integrity of American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 components. The content of HNP TS 3/4.4.10 is as follows:

LIMITING CONDITION FOR OPERATION

3.4.10 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Specification 4.4.10.

APPLICABILITY: All MODES.

ACTION:

- a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.
- b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.
- c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

SURVEILLANCE REQUIREMENTS

4.4.10 Each reactor coolant pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975. In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals.

Administrative Control TS 6.1.2

Administrative Control TS 6.1.2 addresses the responsibility of the Superintendent-Shift Operations for the control room command function. The content of TS 6.1.2 is as follows:

- 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.

TS Table 4.3-2 – Engineered Safety Features Actuation System Instrumentation Surveillance Requirements

TS 3/4.3.2 addresses the operability of the Engineered Safety Features Actuation System (ESFAS) instrumentation and interlocks, ensuring that: (1) the associated ACTION will be initiated when the parameter monitored by each channel or combination thereof reaches its Setpoint; (2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance consistent with maintaining an appropriate level of reliability of the ESFAS instrumentation; and (3) sufficient system functional capability is available from diverse parameters.

TS Table 4.3-2 specifies the Surveillance Requirements for demonstrating operability of each ESFAS instrumentation channel and interlock and the automatic actuation logic and relays, with the surveillance frequency as specified in the Surveillance Frequency Control Program (SFCP). In addition to being specified by the SFCP, some of the slave relay test frequencies are impacted by Note 3 of the table. This Note reads as follows:

- (3) Except for relays K601, K602, K603, K608, K610, K615, K616, K617, K622, K636, K739, K740 and K741 which shall be tested at the frequency specified in the Surveillance Frequency Control Program and during each COLD SHUTDOWN exceeding 72 hours unless they have been tested within the previous 92 days.

2.2 Reason for the Proposed Change

2.2.1 Operating License Section 2.G

Section 2.G of the HNP Facility Operating License requires violations of the conditions included in Section 2.C of the Facility Operating License to be reported to the NRC. While the threshold for some of the conditions included in Section 2.C duplicate those defined in regulations 10 CFR 50.72, "Immediate notification requirements for operating nuclear power reactors," and 10 CFR 50.73, "Licensee event report system," the requirements in HNP Facility Operating License Section 2.G have different deadlines than those defined in the regulations following the changes to the reporting regulations that became effective in January 2001 (65 FR 63769, October 25, 2000). This difference in reporting requirements has led to variations in reporting since many facility operating licenses do not contain the subject condition and has decreased the benefits of the rulemaking for HNP.

An operating license improvement was announced in the *Federal Register* on November 4, 2005 (70 FR 67202) to eliminate the license condition requiring reporting of violations of Section 2.C as part of the consolidated line item improvement process (CLIP). The model safety evaluation (SE) and a model no significant hazards consideration (NSHC) determination relating to the elimination of this license condition were issued in a *Federal Register* notice on August 29, 2005 (70 FR 51098).

2.2.2 Technical Specifications

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

(ISTS) requirements (i.e., NUREG-1431 for Westinghouse Plants). As discussed in Section 16.0, Revision 3, "Technical Specifications," dated March 2010 (ADAMS Accession No. ML100351425), of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: Light Water Reactor (LWR) Edition," TS change requests for facilities with TS based on previous standard TS should comply with comparable provisions in current ISTS NUREGs to the extent possible or justify deviations from the ISTS. The proposed changes found in this license amendment request are generally consistent with the requirements in the current ISTS.

Technical Specifications Index

The Index does not meet the criteria specified in 10 CFR 50.36 requiring its inclusion within TS. The proposed change would remove the Index from the TS and transfer the responsibility for maintenance and issuance of updates to the TS Index to licensee control. Placing the Index under licensee control eliminates the regulatory burden of submitting Index pages for NRC review and allows timely administrative corrections and improvements to the Index without NRC review and approval.

TS 3/4.4.10 – Structural Integrity

By letter dated June 12, 2008 (ADAMS Accession No. ML081430111), the NRC issued License Amendment No. 127 to the HNP Facility Operating License that allowed for the deletion of HNP TS SR 4.0.5, which had previously established the surveillance requirements for inservice inspection and testing of ASME Class 1, 2, and 3 components. The removal of this SR did not eliminate any inservice inspections and did not relinquish HNP of its responsibility to seek relief from Code inspection requirements when they are impractical. Instead, this change eliminated the redundancy of the ISI requirements from the TS since they are already covered under 10 CFR 50.55a.

HNP TS 3/4.4.10 is also redundant to the requirements contained within 10 CFR 50.55a. Per the Bases for HNP TS 3/4.4.10:

The inservice inspection (ISI) and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

With this proposed change, the structural integrity of ASME Code Class 1, 2 and 3 components will continue to be maintained by compliance with 10 CFR 50.55a, as implemented through the HNP ISI Program. Additionally, the requirements of SR 4.4.10 will be relocated to a new program in the Administrative Controls section of TS. This is an administrative change since the relocation does not alter the current inspection requirements for the reactor coolant pump flywheels.

Administrative Control TS 6.1.2

The proposed change would eliminate the annual management directive regarding the responsibility of the control room command function. This directive is redundant to the requirement imposed per HNP TS Table 6.2-1, "Minimum Shift Crew Composition," which states:

During any absence of the Superintendent – Shift Operations from the control room while the unit is in MODE 1, 2, 3, or 4, an individual (other than the Shift Technical Advisor) with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Superintendent – Shift Operations from the control room while the unit is in MODE 5 or 6, an individual with a valid Senior Operator license or Operator license shall be designated to assume the control room command function.

TS 3/4.3.2 Table 4.3-2 Note 3

The NRC previously approved of the SFCP for use by HNP per License Amendment No. 154 by letter dated November 29, 2016 (ADAMS Accession No. ML16200A285). As noted in Section 2.1.2 above, Note (3) of HNP TS Table 4.3-2 allows for the utilization of the SFCP to establish the surveillance frequency of the slave relay tests performed for relays K601, K602, K603, K608, K610, K615, K616, K617, K622, K636, K739, K740 and K741. However, there is the additional caveat that the relays shall be tested during each COLD SHUTDOWN exceeding 72 hours unless they have been tested within the previous 92 days. This caveat restricts the ability to extend the surveillance frequency beyond the previously established frequency of a refueling cycle (i.e., 18 months), despite the relays being within the purview of the SFCP. In order to be able to utilize the SFCP to extend the surveillance frequency for these relays, Note (3) will need to be revised to remove this restriction.

2.3 Description of the Proposed Change

Operating License Section 2.G

Consistent with the CLIIP Notice of Availability (70 FR 67202), the proposed amendment will delete Section 2.G of Facility Operating License NPF-63. In place of the current content outlined in Section 2.1.1 above, Section 2.G of the operating license will state "Deleted."

Technical Specifications Index

Following the approval of this license amendment request, responsibility for maintenance and issuance of updates to the TS Index will transfer from the NRC to Duke Energy. The Index will no longer be included in the NRC-issued TS and will no longer be part of the TS (Appendix A to the Facility Operating License).

TS 3/4.4.10 – Structural Integrity

The text for TS 3/4.4.10 will be deleted in its entirety and existing SR 4.4.10 will be relocated to a new program, "Reactor Coolant Pump Flywheel Inspection Program," as TS 6.8.4.s in the Administrative Controls section of TS. The new program will read as follows:

s. Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals.

This is consistent with ISTS 5.5.7, "Reactor Coolant Pump Flywheel Inspection Program," of NUREG-1431.

Administrative Control TS 6.1.2

The text for TS 6.1.2 will be revised to remove the management directive requirement as follows:

- 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.~~

TS 3/4.3.2 Table 4.3-2 Note 3

The text for TS Table 4.3-2 Note 3 would be revised as follows (new text in bold):

- (3) ~~Deleted. Except for relays K601, K602, K603, K608, K610, K615, K616, K617, K622, K636, K739, K740 and K741 which shall be tested at the frequency specified in the Surveillance Frequency Control Program and during each COLD SHUTDOWN exceeding 72 hours unless they have been tested within the previous 92 days.~~

Additionally, the surveillance frequency information in Table 4.3-2 for slave relay tests of the following Channel Functional Units will be updated to reflect the removal of Note 3:

- 1.b. Safety Injection: Automatic Actuation Logic and Actuation Relays
- 3.a.2. Containment Isolation: Phase "A" Isolation – Automatic Actuation Logic and Actuation Relays
- 6.g. Auxiliary Feedwater: Steam Line Differential Pressure – High
- 7.a. Safety Injection Switchover to Containment Sump: Automatic Actuation Logic and Actuation Relays
- 7.b. Safety Injection Switchover to Containment Sump: RWST Level – Low-Low
- 8.a. Containment Spray Switchover to Containment Sump: Automatic Actuation Logic and Actuation Relays

3.0 TECHNICAL EVALUATION

Operating License Condition 2.G

Duke Energy has reviewed the model SE published on August 29, 2005, as part of the CLIP Notice of Opportunity to Comment and concluded that the justifications presented in the SE prepared by the NRC staff are applicable to HNP. Per this notice, the NRC states:

For those cases where the current Facility Operating License requirement to report violations is also reportable in accordance with the regulations defined in 10 CFR 50.72 and 10 CFR 50.73, the NRC staff finds that the regulations adequately address this issue and the elimination of the duplicative requirement in the Facility Operating License is acceptable.

Some of the conditions addressed in Section 2.[C] of the Facility Operating License may address the maintenance of particular programs, administrative requirements, or other matters where a violation of the requirement would not result in a report to the NRC in accordance with 10 CFR 50.72 or 10 CFR 50.73. In most cases, there are requirements for reports to the NRC related to these conditions in other regulations, the specific license condition or technical specification, or an NRC-approved program document. In other cases, there are reports to other agencies or news releases that would prompt a report to the NRC (in accordance with 10 CFR 50.72(b)(2)(xi)). The NRC staff also assessed violations of administrative requirements that could be reportable under the current License Condition but that may not have a duplicative requirement in a regulation or other regulatory requirement. The NRC staff finds that the requirements to report such problems within 24 hours with written reports to follow using the LER process is not needed. The NRC staff is confident that the information related to such violations that is actually important to the NRC's regulatory functions would come to light in a time frame comparable to the 60-day LER requirements. The information would become available to the appropriate NRC staff through the inspection program, updates to program documents, resultant licensing actions, public announcements, or some other reliable mechanism.

Therefore, the elimination of Section 2.G of the HNP Facility Operating License will not result in a loss of information to the NRC that would adversely affect either its goal to protect public health and safety or its ability to carry out its various other regulatory responsibilities.

TS Index

The HNP Index is a list which provides information (i.e., page numbers) that aids in locating specific TS sections and associated TS Bases. The Index does not contain any technical information.

The Index does not meet the criteria specified in 10 CFR 50.36 that would require its inclusion within the HNP TS. Specifically, 10 CFR 50.36(b) states the following:

Each license authorizing operation of a production or utilization facility of a type described in § 50.21 or § 50.22 will include technical specifications. The technical

specifications will be derived from the analyses and evaluation included in the safety analysis report, and amendments thereto, submitted pursuant to § 50.34.

Furthermore, 10 CFR 50.36(c) lists the following categories of items to be included in TS:

1. Safety limits, limiting safety system settings, and limiting control settings
2. Limiting conditions for operation
3. Surveillance requirements
4. Design features
5. Administrative controls
6. Decommissioning
7. Initial notification
8. Written Reports

In reviewing 10 CFR 50.36, the Index was found to not fall under any of the categories included in 10 CFR 50.36(c). While 10 CFR 50.36(c)(5) describes Administrative controls as 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, the Index is not relied upon to assure safe operation of the facility.

Additionally, 10 CFR 50.36(a) indicates that a license application may provide other information associated with the TS, like a summary statement of the bases or reasons for such specifications (i.e., the TS Bases), but this additional information shall not become part of the technical specification. Like the TS Bases, the Index does not meet the criteria specified in 10 CFR 50.36 for inclusion within HNP's TS.

An Index for the TS will be maintained under Duke Energy control. The Index will be issued by Duke Energy in conjunction with the implementation of NRC-approved TS amendments that alter the Index.

TS 3/4.4.10 – Structural Integrity

Regulation 10 CFR 50.36(c)(2)(ii) contains the requirements for items that must be in TS, including the four criteria that can be used in the determination of the requirements that must be included in the TS. Items that do not meet any of the four criteria can be relocated from TS to a licensee-controlled document, in which the licensee is able to change the relocated requirements, if necessary, in accordance with 10 CFR 50.59. The four criteria are:

- | | |
|-------------|---|
| Criterion 1 | Installed instrumentation that is used to detect, and indicated in the control room, a significant abnormal degradation of the reactor coolant pressure boundary. |
| Criterion 2 | A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. |
| Criterion 3 | A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. |

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

The requirements contained in HNP TS 3/4.4.10 were evaluated to determine whether they meet any of the 10 CFR 50.36(c)(2)(ii) criteria for items that must be in TS.

As it relates to Criterion 1, HNP TS 3/4.4.10 does not involve installed instrumentation that is used to detect, and indicate in the control room, a significant degradation of the reactor coolant pressure boundary, but rather addresses the structural integrity of ASME Code Class 1, 2, and 3 components. Consequently, TS 3/4.4.10 does not meet Criterion 1.

As it relates to Criterion 2, structural integrity is neither a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis. While HNP TS 3/4.4.10 is related to the integrity of ASME Code Class 1, 2, and 3 components, implementation of the HNP ISI program ensures compliance with 10 CFR 55a and maintaining the integrity of these components. This is accomplished through periodic inspections to verify structural integrity instead of specific monitoring and control during plant operations. Therefore, TS 3/4.4.10 does not meet Criterion 2.

As it relates to Criterion 3, ASME Code Class 1, 2, and 3 components that are part of the primary success path and function to mitigate design basis accidents or transients that either assume the failure of, or present a challenge to, the integrity/operability of these components are included in the individual specification that covers these components. HNP TS 3/4.4.10 addresses only the passive pressure boundary function of these components. Additionally, no specific TS-related structure, system, or component (SSC) is being revised or removed from the TS. Each SSC must continue to meet the requirements of 10 CFR 55a as implemented by the HNP ISI program. Therefore, TS 3/4.4.10 does not meet Criterion 3.

As it relates to Criterion 4, the requirements covered by HNP TS 3/4.4.10 that are being removed have not been shown to be risk significant to public health and safety by either operating experience or probabilistic risk assessment. In addition, the requirements of this TS do not affect the risk review/unavailability monitoring of applicable SSCs. No specific TS-related SSC is being revised or removed from the TS and each SSC must continue to meet the requirements of 10 CFR 55a, as addressed by the HNP ISI program. Therefore, TS 3/4.4.10 does not meet Criterion 4.

Based on the evaluation of HNP TS 3/4.4.10 against the four criteria from 10 CFR 50.36(c)(2)(ii), it was determined that none of the criteria are applicable that would prevent the deletion of TS 3.4.10 and the relocation of SR 4.4.10 to a program in the Administrative Controls section of the HNP TS. The structural integrity of the ASME Code Class 1, 2, and 3 components will continue to be maintained in accordance with the requirements of 10 CFR 50.55a.

Administrative Control TS 6.1.2

The current purpose of HNP TS 6.1.2 is to assign responsibility for the control room command function and direct the issuance of an annual management directive signed by the site vice president to all station personnel regarding this responsibility. Duke Energy is proposing the elimination of the management directive requirement since the content of the directive is

redundant to the requirements already imposed per HNP TS 6.1.2 and TS Table 6.2-1 for the control room command function.

Per Section 10 CFR 50.36(c)(5), the Administrative Controls Section of TS is required to include the provisions relating to organization and management, procedures, record keeping, review and audit, and reporting necessary to ensure safe operation of the facility. The proposed amendment to TS 6.1.2 to remove the management directive requirement does not impact Duke Energy's ability to ensure safe operation of the facility since there is no change related to the responsibility of the control room command function or the method in which it is assigned in the absence of the Superintendent-Shift Operations from the control room.

TS Table 4.3-2, "Engineered Safety Features Actuation System Instrumentation Surveillance Requirements"

Per letter dated November 29, 2016, the NRC issued license amendment No. 154 to the HNP Operating License, allowing for the relocation of specific surveillance frequencies to a licensee-controlled program. In relocating these surveillance frequencies, changes could be made in accordance with the Nuclear Energy Institute document NEI 04-10, "Risk-Informed Technical Specification Initiative 5b, Risk Informed Method for Control of Surveillance Frequencies," Revision 1, as consistent with the NRC-approved Technical Specification Task Force (TSTF) Standard Technical Specifications change TSTF-425, "Relocate Surveillance Frequencies to Licensee Control – RITSTF [Risk Informed TSTF] Initiative 5b," Revision 3 (ADAMS Accession No. ML080280275). The *Federal Register* (FR) notice published on July 6, 2009 (74 FR 31996), announced the availability of the TSTF.

The HNP TS include plant-specific surveillances that are not contained in NUREG-1431 and, therefore, were not included in the NUREG-1431 mark-ups provided in TSTF-425. As identified in the HNP submittal to adopt TSTF-425, the HNP TS are based upon the format and content of the NUREG-0452, "Standard Technical Specifications for Westinghouse Pressurized Water Reactors," series. As a result, the HNP TS surveillance numbers and associated Bases numbers differ from the surveillance numbers and Bases numbers in NUREG-1431, "Standard Technical Specifications – Westinghouse Plants," as shown in TSTF-425, Revision 3. These differences are administrative deviations from TSTF-425 with no impact on the NRC staff's model safety evaluation dated July 6, 2009 (74 FR 31996).

TSTF-425 relocated the 92-day surveillance frequency for performance of ESFAS instrumentation SLAVE RELAY TEST from NUREG-1431 ISTS SR 3.3.2.6 to the SFCP. Per License Amendment No. 154 to the HNP Operating License, the HNP TS SR 4.3.2.1, Table 4.3-2 quarterly ESFAS Instrumentation SLAVE RELAY TEST was similarly relocated to the SFCP. Additionally, the 18-month frequency provided in Note (3) of HNP TS SR 4.3.2.1, Table 4.3-2 for relays K601, K602, K603, K608, K610, K615, K616, K617, K622, K636, K739, K740 and K741 was also relocated to the SFCP. These relays had 18-month frequencies rather than the standard quarterly frequency due to the potential adverse consequences of testing them while the unit was in operation. Furthermore, these relays were, and currently are, required to be tested during any cold shutdown greater than 72 hours unless tested within the previous 92 days. This portion of Note (3) effectively requires performance of the slave relay test for these relays in alignment with the standard quarterly frequency, if the opportunity presents itself (i.e., a cold shutdown greater than 72 hours unless tested within the previous 92 days).

With the relocation of the ESFAS Instrumentation SLAVE RELAY TEST surveillance frequency to the SCFP, this requirement to perform additional surveillances during each COLD SHUTDOWN exceeding 72 hours unless tested within the previous 92 days is no longer necessary. This requirement needlessly restricts the ability to utilize the SFCP to control the surveillance frequency of these relays in accordance with NEI 04-10, Revision 1. Once this extraneous requirement is removed, the means in which slave relay test surveillance frequencies are controlled for ESFAS Instrumentation will align with the industry application of the SFCP for these surveillances.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements and Guidance

10 CFR 50 Appendix A, General Design Criteria 13, 20, 21, and 22

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."

10 CFR 50 Appendix A, GDC 20, "Protection System Functions," states, in relevant part, that, the protection system 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.

10 CFR 50 Appendix A, GDC 21, "Protection System Reliability and Testability," states, in relevant part, that the protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed.

10 CFR 50 Appendix A, GDC 22, "Protection System Independence," states, in relevant part, that the protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in the loss of the protection function or shall be demonstrated to be acceptable on some other defined basis.

10 CFR 50.36, "Technical specifications"

The NRC's regulatory requirements related to the content of the TS are set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, "Technical specifications." This regulation requires that the TS include items in the following five specific categories: (1) safety limits, limiting safety system settings, and limiting control settings, (2) LCOs, (3) Surveillance Requirements, (4) design features, and (5) administrative controls. The regulation does not specify the particular requirements to be included in a plant's TS.

Per 10 CFR 50.36(c)(2)(ii), a TS LCO must be established for each item meeting

one or more of the following criteria:

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

Per 10 CFR 50.36(c)(3), 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.

Per 10 CFR 50.36(c)(5), 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.

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(l) discusses the responsibility of the licensee to designate individuals to be responsible for directing the licensed activities of licensed operators. Additionally, 10 CFR 50.54(m) discusses reactor operators and senior reactor operators licensed under 10 CFR Part 55.

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.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition"

NUREG-0800, Revision 7, Chapter 13, "Conduct of Operations," Section 13.1.2 - 13.1.3, "Operating Organization," provides guidance for the review of the structure, functions, and responsibilities of the onsite organization established to safely operate and maintain the facility.

Conclusion

Duke Energy has evaluated the proposed changes against the applicable regulatory requirements described above. Based on this evaluation, there is reasonable assurance that the health and safety of the public will remain unaffected following the approval of these proposed changes.

4.2 Precedents

Removal of Operating License Condition 2.G.

By letter dated October 17, 2006 (ADAMS Accession No. ML061870078), the NRC previously approved a change to the Facility Operating License for the Seabrook Station, Unit No. 1 to delete License Condition 2.G, "Reporting to the Commission," as described in the Notice of Availability published in the *Federal Register* on April 25, 2006 (71 FR 23955). This change was requested as part of the consolidated line item improvement process and consistent with the model safety evaluation published in the *Federal Register* on November 5, 2005 (70 FR 67202).

Relocation of TS Index to Licensee Control

By letter dated December 18, 2019 (ADAMS Accession No. ML19302E700), the NRC previously approved the removal of the table of contents from the TS and relocation to a licensee-controlled document for the following Exelon Generation Company, LLC facilities: Braidwood Station, Units 1 and 2; Byron Station, Unit Nos. 1 and 2; Dresden Nuclear Power Station, Units 2 and 3; James A. FitzPatrick Nuclear Power Plant; R. E. Ginna Nuclear Power Plant; LaSalle County Station, Units 1 and 2; Limerick Generating Station, Units 1 and 2; Nine Mile Point Nuclear Station, Units 1 and 2; Peach Bottom Atomic Power Station, Units 2 and 3; and Quad Cities Nuclear Power Station, Units 1 and 2.

Removal and Relocation of TS 3/4.4.10 – Structural Integrity

By letter dated August 22, 2011 (ADAMS Accession No. ML111250388), the NRC approved a similar request for Seabrook Station to delete TS 3/4.4.10, "Structural Integrity," and relocate SR 4.4.10, which governs the inspection of the reactor coolant pump flywheel, to the Administrative Controls section of TS.

Removal of TS 6.1.2 Management Directive Requirement

The NRC previously approved similar requests for the units listed below to remove the management requirement to issue an annual directive regarding the control room function. While these other requests addressed management directives issued by the Chief Nuclear Officer (CNO), the justification for the removal of the HNP requirement for the site VP is the same.

- Three Mile Island Nuclear Station, Unit 1 – per letter dated October 5, 2017 (ADAMS Accession No. ML17233A138)
- Turkey Point Nuclear Generating Unit Nos. 3 and 4 – per letter dated March 19, 2018 (ADAMS Accession No. ML18019A078)

- Seabrook Station, Unit No. 1 – per letter dated November 27, 2018 (ADAMS Accession No. ML18247A538)

These amendments also granted approval for other requested changes that are outside the scope of this submittal and are therefore not addressed.

4.3 Significant Hazards Consideration

Pursuant to 10 CFR 50.90, Duke Energy Progress, LLC (Duke Energy), hereby requests a revision to the Shearon Harris Nuclear Power Plant, Unit 1 (HNP), Operating License and Technical Specifications (TS). The proposed amendment would remove License Condition 2.G, "Reporting to the Commission," which requires Duke Energy to report any violations of Operating License Section 2.C within twenty-four hours to the Nuclear Regulatory Commission (NRC) Operations Center via the Emergency Notification System with a written follow-up within 30 days. Additionally, the proposed change would delete HNP TS 3/4.4.10, "Structural Integrity," revise Administrative Control TS 6.1.2 to eliminate the annual management directive requirement, and revise TS Table 4.3-2, "Engineered Safety Features Actuation System Instrumentation Surveillance Requirements," to remove an overly restrictive requirement that impedes the full application of the Surveillance Frequency Control Program for a specific subset of relays. The proposed amendment would also revise HNP TS to remove the Index and place it under licensee control.

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

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

Response: No.

The impacts of the proposed changes do not affect how plant equipment is operated or maintained, nor do the changes impact the intent or substance of the Operating License or TS. No changes are proposed that will make changes to the physical plant or analytical methods.

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

The proposed changes do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes do not alter or prevent the ability of structures, systems or components from performing their intended function to mitigate the consequences on an initiating event with the assumed acceptance limits. The proposed changes do not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. Further, the proposed changes do not increase the types and amounts of radioactive

effluent that may be released offsite, nor significantly increase individual or cumulative occupational or public radiation exposure.

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 neither install or remove any plant equipment, nor alter the design, physical configuration, or mode of operation of any plant structure, system, or component.

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 in this license amendment request do not alter the design, configuration, operation, or function of any plant system, structure, or component. The ability of any operable structure, system, or component to perform its designated safety function is unaffected by these changes. 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 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 upon the above evaluation, Duke Energy concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of "no significant hazards consideration" is justified.

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 CONSIDERATIONS

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.

U.S. Nuclear Regulatory Commission
Serial: RA-20-0252
Enclosure 2

ENCLOSURE 2

PROPOSED TECHNICAL SPECIFICATION CHANGES (MARK-UP)

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

RENEWED LICENSE NUMBER NPF-63

30 PAGES PLUS THE COVER

- (c) The licensee shall maintain appropriate compensatory measures in place until completion of the modifications delineated above.

G. Reporting to the Commission Deleted.

~~Except as otherwise provided in the Technical Specifications or Environmental Protection Plan, Duke Energy Progress, LLC shall report any violations of the requirements contained in Section 2.C of this license in the following manner: initial notification shall be made within twenty four (24) hours to the NRC Operations Center via the Emergency Notification System with written follow up within 30 days in accordance with the procedures described in 10 CFR 50.73 (b), (c) and (e).~~

- H. The licensee shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.
- I. The Updated Safety Analysis Report supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be included in the next scheduled update to the Updated Safety Analysis Report required by 10 CFR 50.71(e)(4) following the issuance of this renewed operating license. Until that update is complete, CP&L* may make changes to the programs and activities described in the supplement without prior Commission approval, provided that CP&L* evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.
- J. The Updated Safety Analysis Report supplement, as revised, describes certain future activities to be completed prior to the period of extended operation. Duke Energy Progress, LLC shall complete these activities no later than October 24, 2026, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.
- K. All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of American Society for Testing and Materials E 185-82 to the extent practicable for the configuration of the specimens in the capsule. Any changes to the capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future inspection. Any changes to storage requirements must be approved by the NRC, as required by 10 CFR Part 50, Appendix H.

*On April 29, 2013, the name "Carolina Power & Light Company" (CP&L) was changed to "Duke Energy Progress, Inc." On August 1, 2015, the name "Duke Energy Progress, Inc." was changed to "Duke Energy Progress, LLC."



INDEX

INDEX

1.0 DEFINITIONS

<u>SECTION</u>	<u>PAGE</u>
1.1 ACTION	1-1
1.2 ACTUATION LOGIC TEST	1-1
1.3 ANALOG CHANNEL OPERATIONAL TEST	1-1
1.4 AXIAL FLUX DIFFERENCE	1-1
1.5 CHANNEL CALIBRATION	1-1
1.6 CHANNEL CHECK	1-1
1.7 CONTAINMENT INTEGRITY	1-2
1.8 CONTROLLED LEAKAGE	1-2
1.9 CORE ALTERATION	1-2
1.9a CORE OPERATING LIMITS REPORT	1-2
1.10 DIGITAL CHANNEL OPERATIONAL TEST	1-2
1.11 DOSE EQUIVALENT I-131	1-2a
1.12 \bar{E} - AVERAGE DISINTEGRATION ENERGY	1-3
1.13 ENGINEERED SAFETY FEATURES RESPONSE TIME	1-3
1.14 EXCLUSION AREA BOUNDARY	1-3
1.15 FREQUENCY NOTATION	1-3
1.16 (DELETED)	1-3
1.17 IDENTIFIED LEAKAGE	1-3
1.17a INSERVICE TESTING PROGRAM	1-3
1.18 MASTER RELAY TEST	1-4
1.19 MEMBER(S) OF THE PUBLIC	1-4
1.20 OFFSITE DOSE CALCULATION MANUAL	1-4
1.21 OPERABLE - OPERABILITY	1-4
1.22 OPERATIONAL MODE - MODE	1-4
1.23 PHYSICS TESTS	1-4
1.24 PRESSURE BOUNDARY LEAKAGE	1-4
1.25 PROCESS CONTROL PROGRAM	1-5
1.26 PURGE - PURGING	1-5
1.27 QUADRANT POWER TILT RATIO	1-5
1.28 RATED THERMAL POWER	1-5
1.29 REACTOR TRIP SYSTEM RESPONSE TIME	1-5
1.30 REPORTABLE EVENT	1-5
1.31 SHUTDOWN MARGIN	1-5

INDEX

DEFINITIONS

<u>SECTION</u>	<u>PAGE</u>
1.32 SITE BOUNDARY	1-5
1.33 SLAVE RELAY TEST	1-6
1.34 (DELETED)	1-6
1.35 SOURCE CHECK	1-6
1.36 STAGGERED TEST BASIS	1-6
1.37 THERMAL POWER	1-6
1.38 TRIP ACTUATING DEVICE OPERATIONAL TEST	1-6
1.39 UNIDENTIFIED LEAKAGE	1-6
1.40 UNRESTRICTED AREA	1-6
1.41 VENTILATION EXHAUST TREATMENT SYSTEM	1-7
1.42 VENTING	1-7
TABLE 1.1 FREQUENCY NOTATION	1-8
TABLE 1.2 OPERATIONAL MODES	1-9

INDEX

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

<u>SECTION</u>	<u>PAGE</u>
<u>2.1 SAFETY LIMITS</u>	
2.1.1 REACTOR CORE	2-1
2.1.2 REACTOR COOLANT SYSTEM PRESSURE	2-1
FIGURE 2.1-1 REACTOR CORE SAFETY LIMITS - THREE LOOPS IN OPERATION	2-2
WITH MEASURED RCS FLOW > [293,540 GPM x (1.0 + C ₁)]	
<u>2.2 LIMITING SAFETY SYSTEM SETTINGS</u>	
2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS	2-1
TABLE 2.2-1 REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS	2-4

BASES

<u>SECTION</u>	<u>PAGE</u>
<u>2.1 SAFETY LIMITS</u>	
2.1.1 REACTOR CORE	B 2-1
2.1.2 REACTOR COOLANT SYSTEM PRESSURE	B 2-2
<u>2.2 LIMITING SAFETY SYSTEM SETTINGS</u>	
2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS	B 2-2

INDEX

3.0/4.0. LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

<u>SECTION</u>	<u>PAGE</u>
<u>3/4.0 APPLICABILITY</u>	3/4 0-1
<u>3/4.1 REACTIVITY CONTROL SYSTEMS</u>	
3/4.1.1 BORATION CONTROL	
Shutdown Margin - MODES 1 and 2	3/4 1-1
Shutdown Margin - MODES 3, 4, and 5	3/4 1-3
FIGURE 3.1-1 (DELETED)	3/4 1-3a
Moderator Temperature Coefficient	3/4 1-4
Minimum Temperature for Criticality	3/4 1-6
3/4.1.2 BORATION SYSTEMS	
Flow Path - Shutdown	3/4 1-7
Flow Paths - Operating	3/4 1-8
Charging Pump - Shutdown	3/4 1-9
Charging Pumps - Operating	3/4 1-10
Borated Water Source - Shutdown	3/4 1-11
Borated Water Sources - Operating	3/4 1-12
3/4.1.3 MOVABLE CONTROL ASSEMBLIES	
Group Height	3/4 1-14
TABLE 3.1-1 ACCIDENT ANALYSES REQUIRING REEVALUATION IN THE EVENT OF AN INOPERABLE ROD	3/4 1-16
Position Indication Systems - Operating	3/4 1-17
Position Indication System - Shutdown	3/4 1-18
Rod Drop Time	3/4 1-19
Shutdown Rod Insertion Limit	3/4 1-20
Control Rod Insertion Limits	3/4 1-21
FIGURE 3.1-2 (DELETED)	3/4 1-22

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

<u>SECTION</u>	<u>PAGE</u>
<u>3/4.2 POWER DISTRIBUTION LIMITS</u>	
3/4.2.1 AXIAL FLUX DIFFERENCE.....	3/4 2-1
FIGURE 3.2-1 (DELETED).....	3/4 2-4
3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - $F_Q(X,Y,Z)$	3/4 2-5
FIGURE 3.2-2 (DELETED).....	3/4 2-8
3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F_{\Delta H}(X,Y)$	3/4 2-9
FIGURE 3.2-3 (DELETED).....	3/4 2-10b
3/4.2.4 QUADRANT POWER TILT RATIO.....	3/4 2-11
3/4.2.5 DNB PARAMETERS.....	3/4 2-14
<u>3/4.3 INSTRUMENTATION</u>	
3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION.....	3/4 3-1
TABLE 3.3-1 REACTOR TRIP SYSTEM INSTRUMENTATION.....	3/4 3-2
TABLE 3.3-2 (DELETED).....	3/4 3-9
TABLE 4.3-1 REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS.....	3/4 3-11
3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION.....	3/4 3-16
TABLE 3.3-3 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION.....	3/4 3-18
TABLE 3.3-4 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS.....	3/4 3-28
TABLE 3.3-5 (DELETED).....	3/4 3-37
TABLE 4.3-2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS.....	3/4 3-41
3/4.3.3 MONITORING INSTRUMENTATION Radiation Monitoring For Plant Operations.....	3/4 3-50

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

<u>SECTION</u>	<u>PAGE</u>
TABLE 3.3-6 RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS	3/4 3-51
TABLE 4.3-3 RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS	3/4 3-54
(Deleted)	3/4 3-56
(Deleted)	3/4 3-57
TABLE 3.3-7 (DELETED)	3/4 3-58
TABLE 4.3-4 (DELETED)	3/4 3-59
(Deleted)	3/4 3-60
TABLE 3.3-8 (DELETED)	3/4 3-61
TABLE 4.3-5 (DELETED)	3/4 3-62
Remote Shutdown System	3/4 3-63
TABLE 3.3-9 REMOTE SHUTDOWN SYSTEM	3/4 3-64
TABLE 4.3-6 REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS	3/4 3-65
Accident Monitoring Instrumentation	3/4 3-66
TABLE 3.3-10 ACCIDENT MONITORING INSTRUMENTATION	3/4 3-68
TABLE 4.3-7 ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS	3/4 3-70
TABLE 3.3-11 (DELETED)	3/4 3-73
(Deleted)	3/4 3-74
(Deleted)	3/4 3-75
(Deleted)	3/4 3-82
TABLE 3.3-13 (DELETED)	3/4 3-83
TABLE 4.3-9 (DELETED)	3/4 3-86
3/4.3.4 (Deleted)	3/4 3-89

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

<u>SECTION</u>	<u>PAGE</u>
<u>3/4.4 REACTOR COOLANT SYSTEM</u>	
3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION	
Startup and Power Operation	3/4 4-1
Hot Standby	3/4 4-2
Hot Shutdown	3/4 4-4
Cold Shutdown - Loops Filled	3/4 4-6
Cold Shutdown - Loops Not Filled	3/4 4-7
3/4.4.2 SAFETY VALVES	
Shutdown	3/4 4-8
Operating	3/4 4-9
3/4.4.3 PRESSURIZER	3/4 4-10
3/4.4.4 RELIEF VALVES	3/4 4-11
3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY	3/4 4-13
TABLE 4.4-1 (DELETED)	3/4 4-18
TABLE 4.4-2 (DELETED)	3/4 4-19
3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE	
Leakage Detection Systems	3/4 4-21
Operational Leakage	3/4 4-23
TABLE 3.4-1 REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES	3/4 4-25
3/4.4.7 CHEMISTRY	3/4 4-26
TABLE 3.4-2 REACTOR COOLANT SYSTEM CHEMISTRY LIMITS	3/4 4-27
TABLE 4.4-3 REACTOR COOLANT SYSTEM CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS	3/4 4-28
3/4.4.8 SPECIFIC ACTIVITY	3/4 4-29
FIGURE 3.4-1 (DELETED)	3/4 4-30
TABLE 4.4-4 REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM	3/4 4-31

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

<u>SECTION</u>	<u>PAGE</u>
3/4.4.9 PRESSURE/TEMPERATURE LIMITS	
Reactor Coolant System	3/4 4-33
FIGURE 3.4-2 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO 36 EFPY	3/4 4-35
FIGURE 3.4-3 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE UP TO 36 EFPY	3/4 4-36
TABLE 4.4-5 DELETED	3/4 4-37
TABLE 4.4-6 MAXIMUM COOLDOWN AND HEATUP RATES FOR MODES 4, 5 AND 6 (WITH REACTOR VESSEL HEAD ON)	3/4 4-38
Pressurizer	3/4 4-39
Overpressure Protection Systems	3/4 4-40
FIGURE 3.4-4 MAXIMUM ALLOWED PORV SETPOINT FOR THE LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM	3/4 4-41
3/4.4.10 STRUCTURAL INTEGRITY	3/4 4-43
3/4.4.11 REACTOR COOLANT SYSTEM VENTS	3/4 4-44
 <u>3/4.5 EMERGENCY CORE COOLING SYSTEMS</u>	
3/4.5.1 ACCUMULATORS	3/4 5-1
3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350°F	3/4 5-3
3/4.5.3 ECCS SUBSYSTEMS - T_{avg} LESS THAN 350°F	3/4 5-7
3/4.5.4 REFUELING WATER STORAGE TANK	3/4 5-9

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

<u>SECTION</u>	<u>PAGE</u>
<u>3/4.6 CONTAINMENT SYSTEMS</u>	
3/4.6.1 PRIMARY CONTAINMENT	
Containment Integrity.....	3/4 6-1
Containment Leakage.....	3/4 6-2
Containment Air Locks.....	3/4 6-4
Internal Pressure.....	3/4 6-6
Air Temperature.....	3/4 6-7
Containment Vessel Structural Integrity.....	3/4 6-8
Containment Ventilation System.....	3/4 6-9
3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS	
Containment Spray System.....	3/4 6-11
Spray Additive System.....	3/4 6-12
Containment Cooling System.....	3/4 6-13
3/4.6.3 CONTAINMENT ISOLATION VALVES.....	
TABLE 3.6-1 (DELETED).....	3/4 6-16
3/4.6.4 COMBUSTIBLE GAS CONTROL	
Hydrogen Monitors.....	3/4 6-30
Electric Hydrogen Recombiners.....	3/4 6-31
3/4.6.5 VACUUM RELIEF SYSTEM.....	
3/4 6-32	
 <u>3/4.7 PLANT SYSTEMS</u>	
3/4.7.1 TURBINE CYCLE	
Safety Valves.....	3/4 7-1
TABLE 3.7-1 MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 3 LOOP OPERATION.....	3/4 7-2
TABLE 3.7-2 STEAM LINE SAFETY VALVES PER LOOP.....	3/4 7-3
Auxiliary Feedwater System.....	3/4 7-4
Condensate Storage Tank.....	3/4 7-6
Specific Activity.....	3/4 7-7
TABLE 4.7-1 SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM.....	3/4 7-8
Main Steam Line Isolation Valves.....	3/4 7-9

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

<u>SECTION</u>	<u>PAGE</u>
3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION.....	3/4 7-10
3/4.7.3 COMPONENT COOLING WATER SYSTEM.....	3/4 7-11
3/4.7.4 EMERGENCY SERVICE WATER SYSTEM.....	3/4 7-12
3/4.7.5 ULTIMATE HEAT SINK.....	3/4 7-13
3/4.7.6 CONTROL ROOM EMERGENCY FILTRATION SYSTEM.....	3/4 7-14
3/4.7.7 REACTOR AUXILIARY BUILDING (RAB) EMERGENCY EXHAUST SYSTEM	3/4 7-17
3/4.7.8 SNUBBERS.....	3/4 7-19
FIGURE 4.7-1 (DELETED).....	3/4 7-24
3/4.7.9 SEALED SOURCE CONTAMINATION.....	3/4 7-25
3/4.7.10 (DELETED).....	3/4 7-27
TABLE 3.7-3 (DELETED).....	3/4 7-27
TABLE 3.7-4 (DELETED).....	3/4 7-27
TABLE 3.7-5 (DELETED).....	3/4 7-27
3/4.7.11 (DELETED).....	3/4 7-27
3/4.7.12 (DELETED).....	3/4 7-28
TABLE 3.7-6 (DELETED).....	3/4 7-29
3/4.7.13 ESSENTIAL SERVICES CHILLED WATER SYSTEM.....	3/4 7-30
3/4.7.14 FUEL STORAGE POOL BORON CONCENTRATION.....	3/4 7-31
<u>3/4.8 ELECTRICAL POWER SYSTEMS</u>	
3/4.8.1 A.C. SOURCES	
Operating.....	3/4 8-1
TABLE 4.8-1 (DELETED).....	3/4 8-10
Shutdown.....	3/4 8-11
3/4.8.2 D.C. SOURCES	
Operating.....	3/4 8-12
TABLE 4.8-2 BATTERY SURVEILLANCE REQUIREMENTS.....	3/4 8-14
Shutdown.....	3/4 8-15
3/4.8.3 ONSITE POWER DISTRIBUTION	
Operating.....	3/4 8-16
Shutdown.....	3/4 8-18

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

<u>SECTION</u>	<u>PAGE</u>
3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES	
Containment Penetration Conductor Overcurrent	
Protective Devices	3/4 8-19
TABLE 3.8-1 DELETED	3/4 8-21
Motor-Operated Valves Thermal Overload Protection	3/4 8-39
TABLE 3.8-2 DELETED	3/4 8-40
<u>3/4.9 REFUELING OPERATIONS</u>	
3/4.9.1 BORON CONCENTRATION	3/4 9-1
TABLE 3.9-1 ADMINISTRATIVE CONTROLS TO PREVENT DILUTION DURING REFUELING	3/4 9-2
3/4.9.2 INSTRUMENTATION	3/4 9-3
3/4.9.3 (DELETED)	3/4 9-4
3/4.9.4 CONTAINMENT BUILDING PENETRATIONS	3/4 9-5
3/4.9.5 (DELETED)	3/4 9-6
3/4.9.6 (DELETED)	3/4 9-7
3/4.9.7 (DELETED)	3/4 9-8
3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION	
High Water Level	3/4 9-9
Low Water Level	3/4 9-10
3/4.9.9 CONTAINMENT VENTILATION ISOLATION SYSTEM	3/4 9-11
3/4.9.10 WATER LEVEL - REACTOR VESSEL	3/4 9-12
3/4.9.11 WATER LEVEL - NEW AND SPENT FUEL POOLS	3/4 9-13
3/4.9.12 FUEL HANDLING BUILDING EMERGENCY EXHAUST SYSTEM	3/4 9-14
<u>3/4.10 SPECIAL TEST EXCEPTIONS</u>	
3/4.10.1 SHUTDOWN MARGIN	3/4 10-1
3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS	3/4 10-2
3/4.10.3 PHYSICS TESTS	3/4 10-3
3/4.10.4 REACTOR COOLANT LOOPS	3/4 10-4
3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN	3/4 10-5

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

<u>SECTION</u>		<u>PAGE</u>
<u>3/4.11</u>	<u>RADIOACTIVE EFFLUENTS</u>	
3/4.11.1	LIQUID EFFLUENTS	
	(DELETED).....	3/4 11-1
	(DELETED).....	3/4 11-7
3/4.11.2	GASEOUS EFFLUENTS	
	(DELETED).....	3/4 11-8
	(DELETED).....	3/4 11-15
	(DELETED).....	3/4 11-16
3/4.11.3	(DELETED).....	3/4 11-17
3/4.11.4	(DELETED).....	3/4 11-19
<u>3/4.12</u>	<u>RADIOLOGICAL ENVIRONMENTAL MONITORING</u>	
3/4.12.1	(DELETED).....	3/4 12-1
3/4.12.2	(DELETED).....	3/4 12-1
3/4.12.3	(DELETED).....	3/4 12-1

INDEX

3.0/4.0 BASES

<u>SECTION</u>	<u>PAGE</u>
<u>3/4.0 APPLICABILITY</u>	B 3/4 0-1
<u>3/4.1 REACTIVITY CONTROL SYSTEMS</u>	
3/4.1.1 BORATION CONTROL	B 3/4 1-1
3/4.1.2 BORATION SYSTEMS	B 3/4 1-2
3/4.1.3 MOVABLE CONTROL ASSEMBLIES	B 3/4 1-3
<u>3/4.2 POWER DISTRIBUTION LIMITS</u>	B 3/4 2-1
3/4.2.1 AXIAL FLUX DIFFERENCE	B 3/4 2-1
3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR	B 3/4 2-2a
FIGURE B 3/4.2-1 (DELETED)	B 3/4 2-3
3/4.2.4 QUADRANT POWER TILT RATIO	B 3/4 2-6
3/4.2.5 DNB PARAMETERS	B 3/4 2-6
<u>3/4.3 INSTRUMENTATION</u>	
3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM INSTRUMENTATION AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION	B 3/4 3-1
3/4.3.3 MONITORING INSTRUMENTATION	B 3/4 3-3
3/4.3.4 (DELETED)	B 3/4 3-6
<u>3/4.4 REACTOR COOLANT SYSTEM</u>	
3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION	B 3/4 4-1
3/4.4.2 SAFETY VALVES	B 3/4 4-1
3/4.4.3 PRESSURIZER	B 3/4 4-2
3/4.4.4 RELIEF VALVES	B 3/4 4-2
3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY	B 3/4 4-2b
3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE	B 3/4 4-3
3/4.4.7 CHEMISTRY	B 3/4 4-4
3/4.4.8 SPECIFIC ACTIVITY	B 3/4 4-5
3/4.4.9 PRESSURE/TEMPERATURE LIMITS	B 3/4 4-6

INDEX

BASES

<u>SECTION</u>	<u>PAGE</u>
TABLE B 3/4.4-1 REACTOR VESSEL TOUGHNESS	B 3/4 4-8
FIGURE B 3/4.4-1 FAST NEUTRON FLUENCE (E>1MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE	B 3/4 4-9
FIGURE B 3/4.4-2 (DELETED)	B 3/4 4-10
3/4.4.10 STRUCTURAL INTEGRITY	B 3/4 4-15
3/4.4.11 REACTOR COOLANT SYSTEM VENTS	B 3/4 4-15
 <u>3/4.5 EMERGENCY CORE COOLING SYSTEMS</u>	
3/4.5.1 ACCUMULATORS	B 3/4 5-1
3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS	B 3/4 5-1
3/4.5.4 REFUELING WATER STORAGE TANK	B 3/4 5-2
 <u>3/4.6 CONTAINMENT SYSTEMS</u>	
3/4.6.1 PRIMARY CONTAINMENT	B 3/4 6-1
3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS	B 3/4 6-3
3/4.6.3 CONTAINMENT ISOLATION VALVES	B 3/4 6-4
3/4.6.4 COMBUSTIBLE GAS CONTROL	B 3/4 6-4
3/4.6.5 VACUUM RELIEF SYSTEM	B 3/4 6-4

INDEX

BASES

<u>SECTION</u>	<u>PAGE</u>
----------------	-------------

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE.....	B 3/4 7-1
3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION.....	B 3/4 7-2
3/4.7.3 COMPONENT COOLING WATER SYSTEM.....	B 3/4 7-3
3/4.7.4 EMERGENCY SERVICE WATER SYSTEM.....	B 3/4 7-3
3/4.7.5 ULTIMATE HEAT SINK.....	B 3/4 7-3
3/4.7.6 CONTROL ROOM EMERGENCY FILTRATION SYSTEM.....	B 3/4 7-3
3/4.7.7 REACTOR AUXILIARY BUILDING EMERGENCY EXHAUST SYSTEM.....	B 3/4 7-3a
3/4.7.8 SNUBBERS.....	B 3/4 7-4
3/4.7.9 SEALED SOURCE CONTAMINATION.....	B 3/4 7-5
3/4.7.10 (DELETED).....	B 3/4 7-5
3/4.7.11 (DELETED).....	B 3/4 7-5
3/4.7.12 (DELETED).....	B 3/4 7-5
3/4.7.13 ESSENTIAL SERVICES CHILLED WATER SYSTEM.....	B 3/4 7-5

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1, 3/4.8.2, AND 3/4.8.3 A.C. SOURCES, D.C. SOURCES, AND ONSITE POWER DISTRIBUTION.....	B 3/4 8-1
3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES.....	B 3/4 8-3

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION.....	B 3/4 9-1
3/4.9.2 INSTRUMENTATION.....	B 3/4 9-1
3/4.9.3 (DELETED).....	B 3/4 9-1
3/4.9.4 CONTAINMENT BUILDING PENETRATIONS.....	B 3/4 9-1
3/4.9.5 (DELETED).....	B 3/4 9-2
3/4.9.6 (DELETED).....	B 3/4 9-2
3/4.9.7 (DELETED).....	B 3/4 9-2
3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION.....	B 3/4 9-2
3/4.9.9 CONTAINMENT VENTILATION ISOLATION SYSTEM.....	B 3/4 9-2
3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL AND NEW AND SPENT FUEL POOLS.....	B 3/4 9-3
3/4.9.12 FUEL HANDLING BUILDING EMERGENCY EXHAUST SYSTEM.....	B 3/4 9-4

INDEX

BASES

<u>SECTION</u>	<u>PAGE</u>
<u>3/4.10 SPECIAL TEST EXCEPTIONS</u>	
3/4.10.1 SHUTDOWN MARGIN	B 3/4 10-1
3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS	B 3/4 10-1
3/4.10.3 PHYSICS TESTS.....	B 3/4 10-1
3/4.10.4 REACTOR COOLANT LOOPS.....	B 3/4 10-1
3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN	B 3/4 10-1
<u>3/4.11 RADIOACTIVE EFFLUENTS</u>	
3/4.11.1 (DELETED).....	B 3/4 11-1
3/4.11.2 (DELETED).....	B 3/4 11-1
3/4.11.3 (DELETED).....	B 3/4 11-2
3/4.11.4 (DELETED).....	B 3/4 11-2
<u>3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING</u>	
3/4.12.1 (DELETED).....	B 3/4 12-1
3/4.12.2 (DELETED).....	B 3/4 12-1
3/4.12.3 (DELETED).....	B 3/4 12-1

INDEX

5.0 DESIGN FEATURES

<u>SECTION</u>	<u>PAGE</u>
<u>5.1 SITE</u>	
5.1.1 EXCLUSION AREA	5-1
5.1.2 LOW POPULATION ZONE	5-1
5.1.3 MAP DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS	5-1
FIGURE 5.1-1 EXCLUSION AREA	5-2
FIGURE 5.1-2 LOW POPULATION ZONE	5-3
FIGURE 5.1-3 SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS	5-4
FIGURE 5.1-4 ROUTINE GASEOUS RADIOACTIVE EFFLUENT RELEASE POINTS	5-5
<u>5.2 CONTAINMENT</u>	
5.2.1 CONFIGURATION	5-1
5.2.2 DESIGN PRESSURE AND TEMPERATURE	5-6
<u>5.3 REACTOR CORE</u>	
5.3.1 FUEL ASSEMBLIES	5-6
5.3.2 CONTROL ROD ASSEMBLIES	5-6
<u>5.4 REACTOR COOLANT SYSTEM</u>	
5.4.1 DESIGN PRESSURE AND TEMPERATURE	5-6
5.4.2 VOLUME	5-6
<u>5.5 METEOROLOGICAL TOWER LOCATION</u>	5-6a
<u>5.6 FUEL STORAGE</u>	
5.6.1 CRITICALITY	5-7
5.6.2 DRAINAGE	5-7a
5.6.3 CAPACITY	5-7a
<u>5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT</u>	5-7b
FIGURE 5.6-1 POOLS "C" and "D" BURNUP VERSUS ENRICHMENT FOR PWR FUEL	5-7c
FIGURE 5.6-2 POOLS "A" and "B" BURNUP VERSUS ENRICHMENT FOR PWR FUEL	5-7d
TABLE 5.7-1 COMPONENT CYCLIC OR TRANSIENT LIMITS	5-8

INDEX

6.0 ADMINISTRATIVE CONTROLS

<u>SECTION</u>	<u>PAGE</u>
<u>6.1 RESPONSIBILITY</u>	6-1
<u>6.2 ORGANIZATION</u>	6-1
6.2.1 ONSITE AND OFFSITE ORGANIZATION	6-1
6.2.2 UNIT STAFF	6-1a
FIGURE 6.2-1 DELETED	6-3
FIGURE 6.2-2 DELETED	6-4
TABLE 6.2-1 MINIMUM SHIFT CREW COMPOSITION	6-5
6.2.3 DELETED	6-6
6.2.4 SHIFT TECHNICAL ADVISOR	6-6
<u>6.3 DELETED</u>	6-6
<u>6.4 TRAINING</u>	6-7
<u>6.5 DELETED</u>	6-7

INDEX

ADMINISTRATIVE CONTROLS

<u>SECTION</u>	<u>PAGE</u>
<u>6.6 REPORTABLE EVENT ACTION</u>	6-16
<u>6.7 SAFETY LIMIT VIOLATION</u>	6-16
<u>6.8 PROCEDURES AND PROGRAMS</u>	6-16
<u>6.9 REPORTING REQUIREMENTS</u>	
6.9.1 ROUTINE REPORTS.....	6-20
Annual Radiological Environmental Operating Report.....	6-21
Annual Radioactive Effluent Release Report.....	6-22
Core Operating Limits Report.....	6-24
Steam Generator Tube Inspection Report	6-24c
6.9.2 SPECIAL REPORTS.....	6-24
<u>6.10 DELETED</u>	6-24
<u>6.11 RADIATION PROTECTION PROGRAM</u>	6-26
<u>6.12 HIGH RADIATION AREA</u>	6-26
<u>6.13 PROCESS CONTROL PROGRAM (PCP)</u>	6-27

INDEX

ADMINISTRATIVE CONTROLS

<u>SECTION</u>	<u>PAGE</u>
6.14 <u>OFFSITE DOSE CALCULATION MANUAL (ODCM)</u>	6-27
6.15 <u>(DELETED)</u>	6-28

TABLE 4.3-2
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

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

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

<u>CHANNEL FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>ANALOG CHANNEL OPERATIONAL TEST</u>	<u>TRIP ACTUATING DEVICE OPERATIONAL TEST</u>	<u>ACTUATION LOGIC TEST</u>	<u>MASTER RELAY TEST</u>	<u>SLAVE RELAY TEST</u>	<u>MODES FOR WHICH SURVEILLANCE IS REQUIRED</u>
2. Containment Spray								
a. Manual Initiation	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	SFCP(1)	SFCP	1, 2, 3, 4
c. Containment Pressure--High-3	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	N.A.	1, 2, 3
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	SFCP(1)	SFCP (3)	1, 2, 3, 4
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
b. Phase "B" Isolation								
1) Manual Containment Spray Initiation	See Item 2.a. above for Manual Containment Spray Surveillance Requirements.							
2) Automatic Actuation Logic Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	SFCP(1)	SFCP	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

<u>CHANNEL FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>ANALOG CHANNEL OPERATIONAL TEST</u>	<u>TRIP ACTUATING DEVICE OPERATIONAL TEST</u>	<u>ACTUATION LOGIC TEST</u>	<u>MASTER RELAY TEST</u>	<u>SLAVE RELAY TEST</u>	<u>MODES FOR WHICH SURVEILLANCE IS REQUIRED</u>
6. Auxiliary Feedwater (Continued)								
f. Trip of All Main Feedwater Pumps Start Motor-Driven Pumps	N.A.	N.A.	N.A.	SFCP	N.A.	N.A.	N.A.	1, 2
g. Steam Line Differential Pressure--High Coincident With Main Steam Line Isolation (Causes AFW Isolation)	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	SFCP (3)	1, 2, 3
	See Item 4. above for all Main Steam Line Isolation Surveillance Requirements.							
7. Safety Injection Switchover to Containment Sump								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	SFCP(1)	SFCP (3)	1, 2, 3, 4
b. RWST Level --Low-Low Coincident With Safety Injection	SFCP	SFCP	SFCP	N.A.	N.A.	N.A.	SFCP (3)	1, 2, 3, 4
	See Item 1. above for all Safety Injection Surveillance Requirements.							
8. Containment Spray Switchover to Containment Sump								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	SFCP(1)	SFCP(1)	SFCP (3)	1, 2, 3, 4

TABLE 4.3-2 (Continued)

TABLE NOTATION

- (1) Each train shall be tested at the frequency specified in the Surveillance Frequency Control Program.
 - (2) The Surveillance Requirements of Specification 4.9.9 apply during CORE ALTERATIONS or movement of irradiated fuel in containment.
 - (3) ~~Except for relays K601, K602, K603, K608, K610, K615, K616, K617, K622, K636, K739, K740 and K741 which shall be tested at the frequency specified in the Surveillance Frequency Control Program and during each COLD SHUTDOWN exceeding 72 hours unless they have been tested within the previous 92 days.~~ Deleted.
 - (4) The Steam Line Isolation-Safety Injection (Block-Reset) switches enable the Negative Steam Line Pressure Rate--High signal (item 4.e) when used below the P-11 setpoint. Verify proper operation of these switches each time they are used.
- * Setpoint verification not required.
- # During CORE ALTERATIONS or movement of irradiated fuel in containment.
- ** Trip Function automatically blocked above P-11 and may be blocked below P-11 when safety injection or low steamline pressure is not blocked.

REACTOR COOLANT SYSTEM

3/4.4.10 STRUCTURAL INTEGRITY

ADD:
- DELETED

~~LIMITING CONDITION FOR OPERATION~~

~~3.4.10 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Specification 4.4.10.~~

~~APPLICABILITY: All MODES.~~

~~ACTION:~~

- ~~a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.~~
- ~~b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.~~
- ~~c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.~~

~~SURVEILLANCE REQUIREMENTS~~

~~4.4.10 Each reactor coolant pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975. In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals.~~

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

- 6.1.1 The plant 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 health 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.

ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

p. Surveillance Frequency Control Program

This program provides controls for Surveillance Frequencies. The program shall ensure that Surveillance Requirements specified in the Technical Specifications are performed at intervals sufficient to assure the associated Limiting Conditions for Operation are met.

- a. The Surveillance Frequency Control Program shall contain a list of Frequencies of those Surveillance Requirements for which the Frequency is controlled by the program.
- b. Changes to the Frequencies listed in the Surveillance Frequency Control Program shall be made in accordance with NEI 04-10, "Risk-Informed Method for Control of Surveillance Frequencies," Revision 1.
- c. The provisions of Surveillance Requirements 4.0.2 and 4.0.3 are applicable to the Frequencies established in the Surveillance Frequency Control Program.

q. Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

1. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:
 - a. An API gravity or an absolute specific gravity within limits,
 - b. A flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and
 - c. A clear and bright appearance with proper color or a water and sediment content within limits.
2. Within 31 days following addition of the new fuel oil to storage tanks, verify that the properties of the new fuel oil, other than those addressed in 1., above, are within limits for ASTM 2D fuel oil, and
3. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days.

The provisions of Surveillance Requirement 4.0.2 and Surveillance Requirement 4.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

Add INSERT 1 from the next page.

INSERT 1

s. Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

In lieu of Position C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals.

U.S. Nuclear Regulatory Commission
Serial: RA-20-0252
Enclosure 3

ENCLOSURE 3

PROPOSED TECHNICAL SPECIFICATION BASES CHANGES (MARK-UP)

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

RENEWED LICENSE NUMBER NPF-63

1 PAGE PLUS THE COVER

REACTOR COOLANT SYSTEM

BASES

LOW TEMPERATURE OVERPRESSURE PROTECTION (Continued)

of the LTOPS assuming various mass input and heat input transients. Operation with a PORV setpoint less than or equal to the maximum setpoint ensures that Appendix G criteria will not be violated with consideration for a maximum pressure overshoot beyond the PORV setpoint which can occur as a result of time delays in signal processing and valve opening, instrument uncertainties, and single failure. LTOP instrument uncertainties are controlled by the Technical Specification Equipment List Program, Plant Procedure PLP-106. To ensure that mass and heat input transients more severe than those assumed cannot occur, Technical Specifications require lockout of all but one charging/safety injection pump while in MODES 4 (below 325°F), 5 and 6 with the reactor vessel head installed and disallow start of an RCP if secondary temperature is more than 50°F above primary temperature.

The maximum allowed PORV setpoint for the LTOPS will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50, Appendix H, and the reactor vessel service life.

3/4.4.10 STRUCTURAL INTEGRITY

- Deleted per Amendment No. ____

~~The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).~~

~~Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1977 Edition and Addenda through Summer 1978.~~

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensable gases and/or steam from the Reactor Coolant System that could inhibit natural circulation core cooling. The OPERABILITY of at least one Reactor Coolant System vent path from the reactor vessel head and the pressurizer steam space ensures that the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vents are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plant Requirements," November 1980.