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10 CFR 50.90

June 7, 2021
Serial: RA-21-0112

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 Regarding Administrative Change to Reflect Development of a Technical Requirements Manual

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), Technical Specifications (TS). The proposed amendment will revise HNP TS to reflect the transition of the licensee-controlled plant procedure PLP-106, “Technical Specification Equipment List Program,” to a licensee-controlled Technical Requirements Manual (TRM). The proposed change is an administrative change to the HNP TS with no impact on technical content.

The Enclosure provides a description and assessment of the proposed change. Attachments 1 and 2 provide copies of the proposed changes to the TS and TS Bases, respectively. Attachment 2 is provided for information only, as changes to the HNP TS Bases will be processed in accordance with HNP TS 6.8.4.n, “Technical Specifications (TS) Bases Control Program,” upon implementation of the amendment.

The proposed change has 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 change involves no significant hazards consideration. Additionally, this letter contains no regulatory commitments.

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

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

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 June 7, 2021.

Sincerely,

Kim E. Maza

Kim E. Maza
Site Vice President
Harris Nuclear Plant

Enclosure: Description and Assessment of the Proposed Change

Attachment 1: Proposed Technical Specification Changes (Mark-up)
Attachment 2: 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

ENCLOSURE

DESCRIPTION AND ASSESSMENT OF THE PROPOSED CHANGE

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

RENEWED LICENSE NUMBER NPF-63

4 PAGES PLUS THE COVER

Description and Assessment of the Proposed Change

License Amendment Request Regarding Administrative Change to Reflect Development of a Technical Requirements Manual

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), Technical Specifications (TS). The proposed amendment will revise HNP TS to reflect the transition of the licensee-controlled plant procedure PLP-106, “Technical Specification Equipment List Program,” to a licensee-controlled Technical Requirements Manual (TRM). The proposed change is an administrative change to the HNP TS with no impact on technical content.

2.0 DETAILED DESCRIPTION

2.1 Current Technical Specification

The following HNP TS refer to plant procedure PLP-106 for content that was previously relocated from the TS:

- TS Table 2.2-1, Reactor Trip System Instrumentation Trip Setpoints, Note 8
- TS 3/4.3.1, Reactor Trip System Instrumentation
- TS Table 3.3-2, Reactor Trip System Instrumentation Response Times
- TS 3/4.3.2, Engineered Safety Features Actuation System Instrumentation
- TS Table 3.3-4, Engineered Safety Features Actuation System Instrumentation Trip Setpoints, Note 2
- TS Table 3.3-5, Engineered Safety Features Response Times
- TS 3/4.4.9, Pressure/Temperature Limits
- TS Figure 3.4-2, Reactor Coolant System Cooldown Limitations – Applicable Up to 55 EFPY
- TS Figure 3.4-3, Reactor Coolant System Heatup Limitations – Applicable Up to 55 EFPY
- TS Table 4.4-5, Reactor Vessel Material Surveillance Program
- TS Figure 3.4-4, Maximum Allowed PORV Setpoint for the Low Temperature Overpressure Protection System
- TS 3/4.6.3, Containment Isolation Valves
- TS Table 3.6-1, Containment Isolation Valves
- TS 3/4.7.8, Snubbers
- TS Figure 4.7-1, Sample Plan (2) for Snubber Functional Test
- TS 3/4.8.4, Electrical Equipment Protective Devices

2.2 Reason for the Proposed Change

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).

HNP has not converted to the Improved Standard TS structure and has continued to maintain a licensee-controlled document in the form of plant procedure PLP-106, titled "Technical Specification Equipment List Program," to capture relocated TS content approved by the NRC, including equipment lists, figures, and surveillance programs. PLP-106 is a document incorporated by reference into the HNP Final Safety Analysis Report (FSAR) and is subject to the update and reporting requirements of 10 CFR 50.71(e) and change controls of 10 CFR 50.59.

To standardize the process of updating the licensee-controlled document with that of the rest of the Duke Energy fleet, PLP-106 is being converted into a TRM. As a TRM, the content previously contained in PLP-106 will continue to be incorporated by reference in the FSAR and subject to the update and reporting requirements of 10 CFR 50.71(e), with changes processed in accordance with 10 CFR 50.59, without the required additional resources associated with processing a change to a plant procedure.

2.3 Description of the Proposed Change

The proposed administrative change will replace all TS references to plant procedure PLP-106, as listed in Section 2.1 above, with references to the TRM.

3.0 TECHNICAL EVALUATION

The proposed change to reference a TRM in place of plant procedure PLP-106 does not adversely alter the current TS or introduce any new TS requirements. The relocated TS content currently captured in PLP-106 will be maintained in the TRM, with changes to the licensee-controlled content processed in accordance with 10 CFR 50.59. When implemented, this change will provide continuity and consistency throughout the Duke Energy fleet in the processing of changes to relocated TS content in licensee-controlled documents. There is no impact on plant operations or systems as a result of the proposed change.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements and Guidance

10 CFR 50.36, "Technical specifications"

The NRC's regulatory requirements related to the content of the TS are set forth in 10 CFR 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) limiting conditions for operation, (3) surveillance requirements, (4) design features, and (5) administrative controls.

Conclusion

Duke Energy has evaluated the proposed change 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 the proposed change.

4.2 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), Technical Specifications (TS). The proposed amendment will revise HNP TS to reflect the transition of the licensee-controlled plant procedure PLP-106, “Technical Specification Equipment List Program,” to a licensee-controlled Technical Requirements Manual (TRM). The proposed change is an administrative change to the HNP TS with no impact on technical content.

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 change involve a significant increase in the probability or consequences of an accident previously evaluated?*

Response: No.

The proposed change to reflect a TRM in place of the currently referenced plant procedure PLP-106 is administrative in nature and does not change the technical content of the TS. The proposed change does not adversely affect accident initiators or precursors nor alter the design assumptions, conditions or configurations of the facility. The proposed change does not alter or prevent the capability of structures, systems and components (SSCs) to perform their intended function to mitigate the consequences of any initiating events within the assumed acceptance limits.

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

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

Response: No.

The proposed change to reflect a TRM in place of the currently referenced plant procedure PLP-106 is administrative in nature and does not change the technical content of the TS. The proposed change does not alter the design requirements of any SSC or its function during accident conditions. The proposed change does not involve a physical alteration to the plant or any changes in methods governing normal plant operation. The proposed change does not alter any assumptions made in the safety analysis.

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

(3) *Does the proposed change involve a significant reduction in a margin of safety?*

Response: No.

The proposed change to reflect a TRM in place of the currently referenced plant procedure PLP-106 is administrative in nature and does not change the technical content of the TS. The proposed change does not alter the way safety limits, limiting safety system settings or limiting conditions for operation are determined. The safety analysis acceptance criteria are not affected by the proposed change. The proposed change will not result in plant operation in a configuration outside the design basis and does not adversely affect systems that respond to safely shutdown the plant and maintain the plant in a safety shutdown condition.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based upon the above evaluation, Duke Energy concludes that the proposed change 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.

4.3 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

The proposed amendment is for administrative, non-technical changes only and would not change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20. The proposed amendment does 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.

ATTACHMENT 1

PROPOSED TECHNICAL SPECIFICATION CHANGES (MARK-UP)

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

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TABLE 2.2-1 (Continued)

TABLE NOTATIONS

The values denoted with [*] are specified in the COLR.

NOTE 3: (Continued)

K_6	=	[*]/°F for T > T" and K_6 = [*] for T ≤ T",
T	=	As defined in Note 1,
T"	=	Reference T_{avg} at RATED THERMAL POWER (≤ [*]°F),
S	=	As defined in Note 1, and
$f_2(\Delta I)$	=	[*].

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.4% of ΔT span for ΔT input, 1.35% of T_{avg} span for T_{avg} input; and 0.6% of ΔI span for ΔI input.

NOTE 5: The sensor error is: 1.3% of ΔT span for $\Delta T/T_{avg}$ temperature measurements; and 0.8% of ΔT span for pressurizer pressure measurements.

NOTE 6: The sensor error (in % span of Steam Flow) is: 1.1% for steam flow; 1.8% for feedwater flow; and 2.4% for steam pressure.

NOTE 7: If the as-found channel setpoint is outside its predefined as-found tolerance, the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

NOTE 8: The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Trip Setpoint in Table 2.2-1 (Nominal Trip Setpoint (NTSP)) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine NTSPs and the as-found and the as-left tolerances are specified in EGR-NGGC-0153, “Engineering Instrument Setpoints.” The as-found and as-left tolerances are specified in ~~PLP-106~~.

the Technical Requirements Manual

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION: As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 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.

4.3.1.2 The REACTOR TRIP SYSTEM RESPONSE TIME of each Reactor trip function shall be verified to be within its limit, specified in the ~~Technical Specification Equipment List Program, plant procedure PLP-106,~~ at the frequency specified in the Surveillance Frequency Control Program.

Technical Requirements Manual

TABLE 3.3-2 REACTOR TRIP SYSTEM INSTRUMENTATION
RESPONSE TIMES

This table is deleted from Technical Specifications.

The information in this table is controlled by the ~~Technical Specification Equipment List Program, plant procedure PLP-106~~.

Technical Requirements Manual

PAGE 3/4 3-10 HAS BEEN DELETED.

SURVEILLANCE REQUIREMENTS

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

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

TABLE 3.3-4 (Continued)

TABLE NOTATIONS

* Time constants utilized in the lead-lag controller for Steam Line Pressure--Low are $\tau_1 \geq$ 50 seconds and $\tau_2 \leq$ 5 seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

** The time constant utilized in the rate-lag controller for Steam Line Pressure-Negative Rate--High is ≥ 50 seconds. CHANNEL CALIBRATION shall ensure that this time constant is adjusted to this value.

The indicated values are the effective, cumulative, rate-compensated pressure drops as seen by the comparator.

NOTE 1: If the as-found channel setpoint is outside its predefined as-found tolerance, the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

NOTE 2: The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Trip Setpoint in Table 3.3-4 (Nominal Trip Setpoint (NTSP)) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine NTSPs and the as-found and the as-left tolerances are specified in EGR-NGGC-0153, "Engineering Instrument Setpoints." The as-found and as-left tolerances are specified in ~~PLP-106~~.

the Technical Requirements Manual

TABLE 3.3-5 ENGINEERED SAFETY FEATURES RESPONSE TIMES

This table is deleted from Technical Specifications.

The information in this table is controlled by the ~~Technical Specification Equipment List Program, plant procedure PLP-106~~.

Technical Requirements Manual

PAGES 3/4 3-38 THROUGH 3/4 3-40 HAVE BEEN DELETED.

LIMITING CONDITION FOR OPERATION

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

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

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

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

Technical Requirements Manual

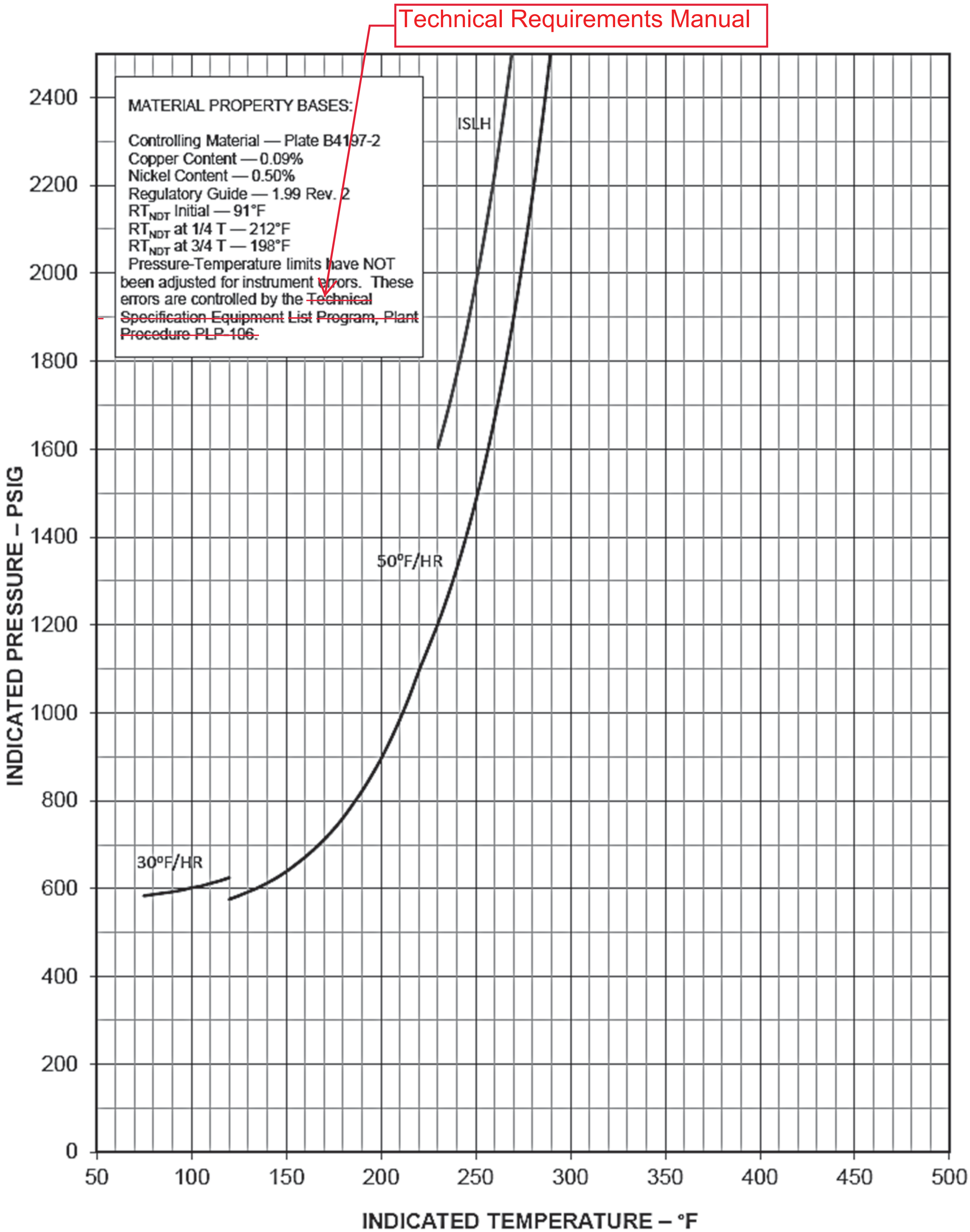


FIGURE 3.4-2
REACTOR COOLANT SYSTEM
COOLDOWN LIMITATIONS – APPLICABLE UP TO 55 EFPY

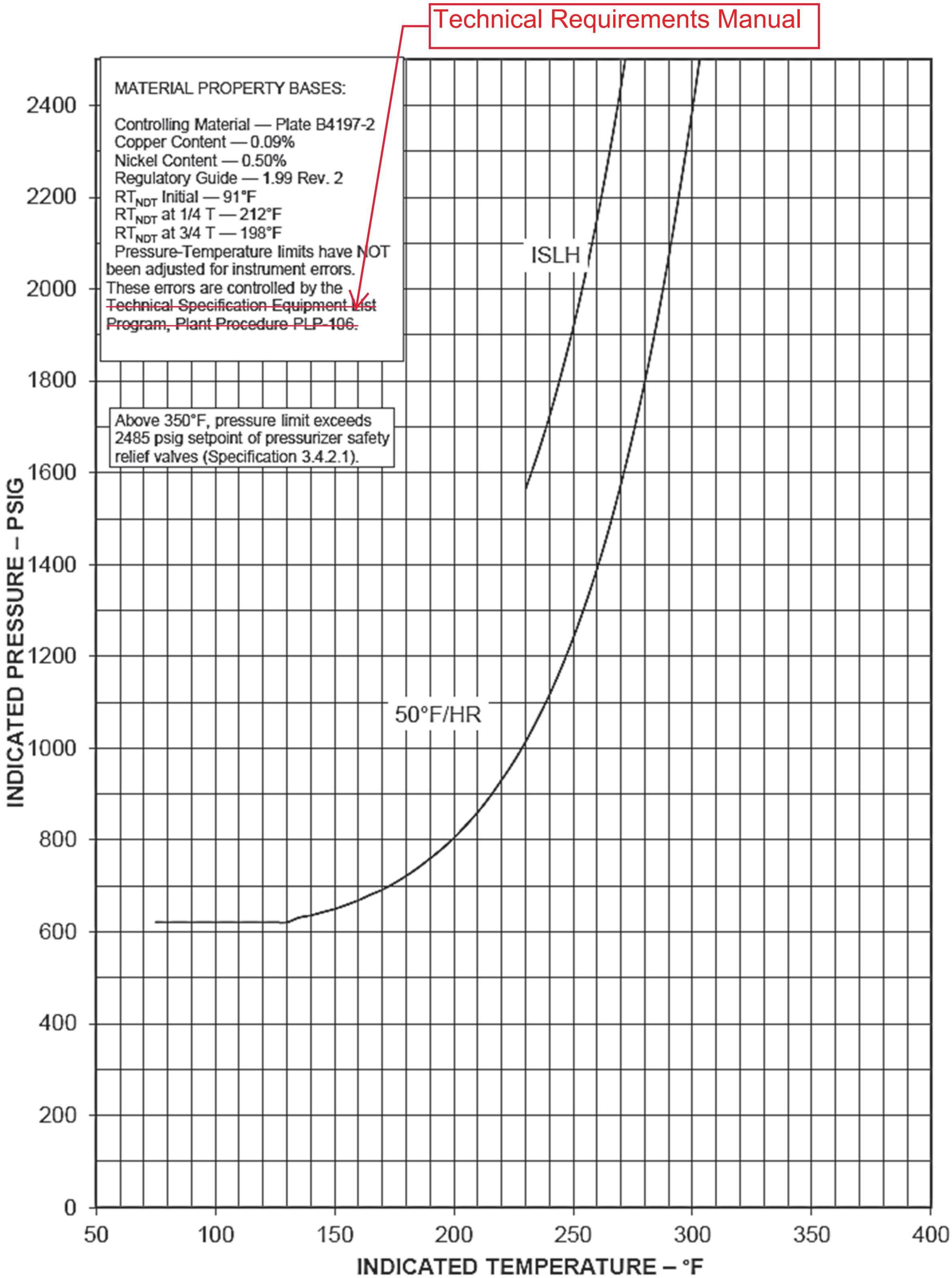


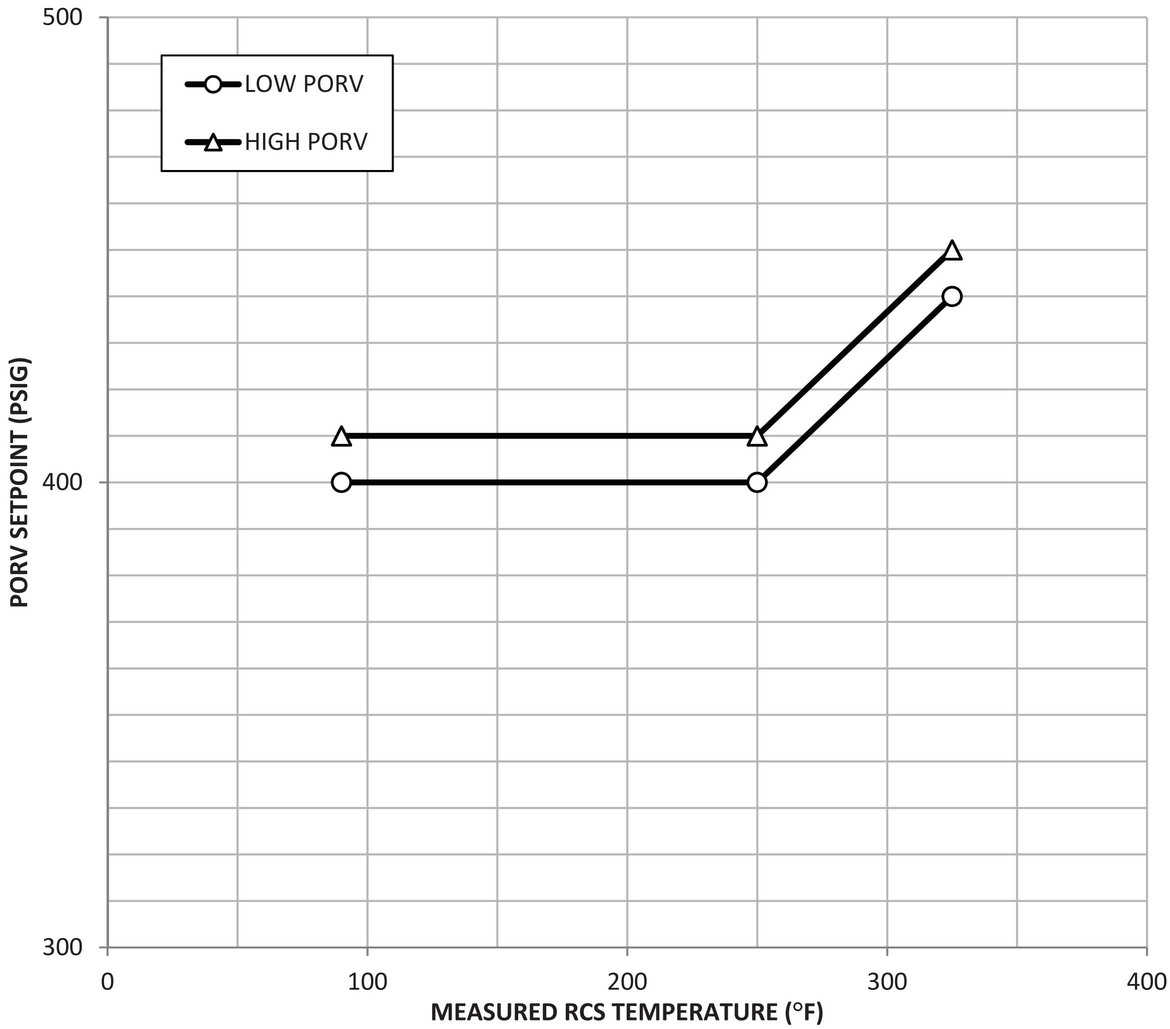
FIGURE 3.4-3
REACTOR COOLANT SYSTEM
HEATUP LIMITATIONS – APPLICABLE UP TO 55 EFPY

TABLE 4.4-5 REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM

This table is deleted from Technical Specifications.

The information in this table is controlled by the ~~Technical Specification Equipment List Program, plant procedure PLP-106~~.

Technical Requirements Manual



Technical Requirements Manual

RCS TEMP (°F)	LOW PORV* (psig)	HIGH PORV* (psig)
90	400	410
250	400	410
325	440	450

* VALUES BASED ON 55 EFPY REACTOR VESSEL DATA

INSTRUMENT ERRORS ARE CONTROLLED BY THE ~~TECHNICAL SPECIFICATION EQUIPMENT LIST PROGRAM, PLANT PROCEDURE PLP-106~~.

FIGURE 3.4-4

MAXIMUM ALLOWED PORV SETPOINT FOR THE LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM

Technical Requirements Manual

LIMITING CONDITION FOR OPERATION

3.6.3 Each containment isolation valve specified in the ~~Technical Specification Equipment List Program, plant procedure PLP-106,~~ shall be OPERABLE with isolation times less than or equal to required isolation times.

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

ACTION:

With one or more of the containment isolation valve(s) inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours or in accordance with the Risk-Informed Completion Time Program, or

b. Isolate each affected penetration within 4 hours or in accordance with the Risk-Informed Completion Time Program by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours or in accordance with the Risk-Informed Completion Time Program by use of at least one closed manual valve or blind flange, or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Each isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test, and verification of isolation time.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.2 Each isolation valve shall be demonstrated OPERABLE at the frequency specified in the Surveillance Frequency Control Program by:

a. Verifying that on a Phase "A" Isolation test signal, each Phase "A" isolation valve actuates to its isolation position;

b. Verifying that on a Phase "B" Isolation test signal, each Phase "B" isolation valve actuates to its isolation position; and

c. Verifying that on a Containment Ventilation Isolation test signal, each normal, preentry purge makeup and exhaust, and containment vacuum relief valve actuates to its isolation position, and

d. Verifying that, on a Safety Injection "S" test signal, each containment isolation valve receiving an "S" signal actuates to its isolation position, and

e. Verifying that, on a Main Steam Isolation test signal, each main steam isolation valve actuates to its isolation position, and

f. Verifying that, on a Main Feedwater Isolation test signal, each feedwater isolation valve actuates to its isolation position.

4.6.3.3 The isolation time of each power-operated or automatic valve shall be determined to be within its limit specified in the ~~Technical Specification Equipment List Program, plant procedure PLP-106,~~ when tested pursuant to the INSERVICE TESTING PROGRAM.

Technical Requirements Manual

TABLE 3.6-1 CONTAINMENT ISOLATION VALVES

This table is deleted from Technical Specifications.

The information in this table is controlled by the ~~Technical Specification Equipment List Program, plant procedure PLP-106~~.

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PAGES 3/4 6-17 THROUGH 3/4 6-29 HAVE BEEN DELETED.

3/4.7.8 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.8 All snubbers shall be OPERABLE. The only snubbers excluded from the requirements are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

APPLICABILITY: MODES 1, 2, 3, and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTION:

With one or more snubbers inoperable on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per augmented inservice inspection program on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.8 Each snubber shall be demonstrated OPERABLE by performance of the augmented inservice inspection program specified in the ~~Technical Specification Equipment List Program, plant procedure PLP-106~~ Technical Requirements Manual.

PAGES 3/4 7-20 THROUGH 3/4 7-23 HAVE BEEN DELETED.

FIGURE 4.7-1 SAMPLE PLAN (2) FOR SNUBBER FUNCTIONAL TEST

This figure is deleted from Technical Specifications and is controlled by the ~~Technical Specification Equipment List Program, plant procedure PLP-106~~.

Technical Requirements Manual

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Specifications 3/4.8.4.1 and 3/4.8.4.2 have been deleted from Technical Specifications and relocated to ~~plant procedure PLP-106~~.

the Technical Requirements Manual.

PAGES 3/4 8-20 THROUGH 3/4 8-43 HAVE BEEN DELETED.

Pages 3/4 8-20, 3/4 8-21, 3/4 8-39, and 3/4 8-40 by Amendment No. 182.

Pages 3/4 8-22 through 3/4 8-38B and 3/4 8-41 through 3/4 8-43 by Amendment No. 13.

ATTACHMENT 2

PROPOSED TECHNICAL SPECIFICATION BASES CHANGES (MARK-UP)

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

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REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

- Pressurizer Water Level – High Setpoint
- Reactor Coolant Flow – Low Setpoint

Notes 7 and 8 have been added to Table 2.2-1 that require verifying both trip setpoint setting as-found and as-left values during surveillance testing. In accordance with 10 CFR 50.36, these functions are Limiting Safety System Settings. Adding test requirements ensures that instruments will function as required to initiate protective systems or actuate mitigating systems at the point assumed in the applicable safety analysis. These notes address NRC staff concerns with Technical Specification Allowable Values. Specifically, calculated Allowable Values may be non-conservative depending upon the evaluation of instrument performance history, and the as-left requirements of the calibration procedures could have an adverse effect on equipment operability. In addition, using Allowable Values as the limiting setting for assessing instrument channel operability may not be fully in compliance with the intent of 10 CFR 50.36, and the existing surveillance requirements would not provide adequate assurance that instruments will always actuate safety functions at the point assumed in the applicable safety analysis. In the Harris Technical Specifications, the term Trip Setpoint is analogous to Nominal Trip Setpoint (NTSP) in TSTF-493.

Note 7 requires a channel performance evaluation when the as-found setting is outside its as-found tolerance. The performance evaluation verifies that the channel will continue to behave in accordance with safety analysis and instrument performance assumptions in the setpoint methodology. The purpose of this evaluation is to provide confidence in the performance prior to returning the channel to service. If the as-found setting is non-conservative with respect to the Allowable Value, the channel is INOPERABLE. If the as-found setting is conservative with respect to the Allowable Value but is outside the as-found tolerance band, the channel is OPERABLE but degraded. The degraded channel condition will be further evaluated during performance of the surveillance. This evaluation will consist of resetting the channel setpoint to within the as-left tolerances applicable to the actual setpoint implemented in the surveillance procedures (field setting), and evaluating the channel response. If the channel is functioning as required and is expected to pass the next surveillance, then the channel is OPERABLE and can be restored to service at the completion of the surveillance. After the surveillance is completed, the channel as-found condition is entered into the corrective action program for further analysis and trending.

Note 8 requires that the as-left channel setting be reset to a value that is within the as-left tolerances about the Trip Setpoint in Table 2.2-1 or within as-left tolerances about a more conservative actual (field) setpoint. As-left channel settings outside the as-left tolerances of ~~PLP-106~~ the Technical Requirements Manual and the surveillance procedures cause the channel to be INOPERABLE.

A tolerance is necessary because no device perfectly measures the process. Additionally, it is not possible to read and adjust a setting to an absolute value due to the readability and/or accuracy of the test instruments or the ability to adjust potentiometers. The as-left tolerance is considered in the setpoint calculation. Failure to set the actual plant trip setpoint to within as-left the tolerances of the NTSP or within as-left tolerances of a more conservative actual field setpoint would invalidate the assumptions in the setpoint calculation, because any subsequent instrument drift would not start from the expected as-left setpoint. The determination will consider whether the instrument is degraded or is capable of being reset and performing its specified safety function. If the channel is determined to be functioning as required (i.e., the channel can be adjusted to within the as-left tolerance and is determined to be functioning normally based on the determination performed prior to returning the channel to service), then the channel is OPERABLE and can be restored to service.

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

If the as-left instrument setting cannot be returned to a setting within the prescribed as-left tolerance band, the instrument would be declared INOPERABLE.

The methodologies for calculating the as-found tolerances and as-left tolerances about the Trip Setpoint or more conservative actual field setpoint are specified in EGR-NGGC-0153, "Engineering Instrument Setpoints," which is incorporated by reference into the FSAR. The actual field setpoint and the associated as-found and as-left tolerances are specified in the Technical Requirements Manual ~~PLP-106~~, the applicable section of which is incorporated by reference into the FSAR.

Limiting Trip Setpoint (LTSP) is generic terminology for the setpoint value calculated by means of the setpoint methodology documented in EGR-NGGC-0153. HNP uses the plant-specific term Nominal Trip Setpoint (NTSP) in place of the generic term LTSP. The NTSP is the LTSP with margin added, and is always equal to or more conservative than the LTSP. The NTSP may use a setting value that is more conservative than the LTSP, but for Technical Specification compliance with 10 CFR 50.36, the plant-specific setpoint term NTSP is cited in Note 8.

The NTSP meets the definition of a Limiting Safety System Setting per 10 CFR 50.36 and is a predetermined setting for a protective channel chosen to ensure that automatic protective actions will prevent exceeding Safety Limits during normal operation and design basis anticipated operational occurrences, and assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The Allowable Value is the least conservative value of the as-found setpoint that the channel can have when tested, such that a channel is OPERABLE if the as-found setpoint is within the as-found tolerance and is conservative with respect to the Allowable Value during a CHANNEL CALIBRATION or CHANNEL OPERATIONAL TEST. As such, the Allowable Value differs from the NTSP by an amount greater than or equal to the expected instrument channel uncertainties, such as drift, during the surveillance interval. In this manner, the actual NTSP setting ensures that a Safety Limit is not exceeded at any given point of time as long as the channel has not drifted beyond expected tolerances during the surveillance interval. Although the channel is OPERABLE under these circumstances, the trip setpoint must be left adjusted to a value within the as-left tolerance band, in accordance with uncertainty assumptions stated in the setpoint methodology (as-left criteria), and confirmed to be operating within the statistical allowances of the uncertainty terms assigned (as-found criteria).

Field setting is the term used for the actual setpoint implemented in the plant surveillance procedures, where margin has been added to the calculated field setting. The as-found and as-left tolerances apply to the field settings implemented in the surveillance procedures to confirm channel performance. A trip setpoint may be set more conservative than the NTSP as necessary in response to plant conditions. However, in this case, the instrument operability must be verified based on the field setting and not the NTSP.

Manual Reactor Trip

The Reactor Trip System includes manual Reactor trip capability.

Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a High and Low Range trip setting. The Low Setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the High Setpoint trip provides protection during power operations to mitigate the consequences of a reactivity excursion from all power levels.

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

the Technical Requirements Manual

Note 2 requires that the as-left channel setting be reset to a value that is within the as-left tolerances about the Trip Setpoint in Table 3.3-4 or within as-left tolerances about a more conservative actual (field) setpoint. As-left channel settings outside the as-left tolerances of ~~PLP-106~~ and the surveillance procedures cause the channel to be INOPERABLE.

A tolerance is necessary because no device perfectly measures the process. Additionally, it is not possible to read and adjust a setting to an absolute value due to the readability and/or accuracy of the test instruments or the ability to adjust potentiometers. The as-left tolerance is considered in the setpoint calculation. Failure to set the actual plant trip setpoint to within as-left the tolerances of the NTSP or within as-left tolerances of a more conservative actual field setpoint would invalidate the assumptions in the setpoint calculation, because any subsequent instrument drift would not start from the expected as-left setpoint. The determination will consider whether the instrument is degraded or is capable of being reset and performing its specified safety function. If the channel is determined to be functioning as required (i.e., the channel can be adjusted to within the as-left tolerance and is determined to be functioning normally based on the determination performed prior to returning the channel to service), then the channel is OPERABLE and can be restored to service. If the as-left instrument setting cannot be returned to a setting within the prescribed as-left tolerance band, the instrument would be declared inoperable.

the Technical Requirements Manual

The methodologies for calculating the as-found tolerances and as-left tolerances about the Trip Setpoint or more conservative actual field setpoint are specified in EGR-NGGC-0153, “Engineering Instrument Setpoints,” which is incorporated by reference into the FSAR. The actual field setpoint and the associated as-found and as-left tolerances are specified in ~~PLP-106~~, the applicable section of which is incorporated by reference into the FSAR.

Limiting Trip Setpoint (LTSP) is generic terminology for the setpoint value calculated by means of the setpoint methodology documented in EGR-NGGC-0153. HNP uses the plant-specific term NTSP in place of the generic term LTSP. The NTSP is the LTSP with margin added, and is always equal to or more conservative than the LTSP. The NTSP may use a setting value that is more conservative than the LTSP, but for Technical Specification compliance with 10 CFR 50.36, the plant-specific setpoint term NTSP is cited in Note 2. The NTSP meets the definition of a Limiting Safety System Setting per 10 CFR 50.36 and is a predetermined setting for a protective channel chosen to ensure that automatic protective actions will prevent exceeding Safety Limits during normal operation and design basis anticipated operational occurrences, and assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The Allowable Value is the least conservative value of the as-found setpoint that the channel can have when tested, such that a channel is OPERABLE if the as-found setpoint is within the as-found tolerance and is conservative with respect to the Allowable Value during a CHANNEL CALIBRATION or CHANNEL OPERATIONAL TEST. As such, the Allowable Value differs from the NTSP by an amount greater than or equal to the expected instrument channel uncertainties, such as drift, during the surveillance interval. In this manner, the actual NTSP setting ensures that a Safety Limit is not exceeded at any given point of time as long as the channel has not drifted beyond expected tolerances during the surveillance interval. Although the channel is OPERABLE under these circumstances, the trip setpoint must be left adjusted to a value within the as-left tolerance band, in accordance with uncertainty assumptions stated in the setpoint methodology (as-left criteria), and confirmed to be operating within the statistical allowances of the uncertainty terms assigned (as-found criteria).

Field setting is the term used for the actual setpoint implemented in the plant surveillance procedures, where margin has been added to the calculated field setting. The as-found and as-left tolerances apply to the field settings implemented in the surveillance procedures to confirm

PRESSURE/TEMPERATURE LIMITS (Continued)

heatup and the time (or coolant temperature) along the heatup ramp. Furthermore, since the thermal stresses at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration.

The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

The composite curves for the heatup rate data and the cooldown rate data in Figures 3.4-2 and 3.4-3 have not been adjusted for possible errors in the pressure and temperature sensing instruments. However, the heatup and cooldown curves in plant operating procedures have been adjusted for these instrument errors. The instrument errors are controlled by ~~the Technical Specification Equipment List Program, Plant Procedure PLP-106~~ the Technical Requirements Manual.

"ISLH" pressure-temperature (P-T) curves may be used for inservice leak and hydrostatic tests with fuel in the reactor vessel. However, ISLH tests required by the ASME code must be completed before the core is critical.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

LOW TEMPERATURE OVERPRESSURE PROTECTION

A Note prohibits application of LCO 3.0.4.b to an inoperable overpressure protection system. There is an increased risk associated with entering MODE 4 from MODE 5 with the overpressure protection system inoperable and the provisions of LCO 3.0.4.b, which allow entry into a MODE or other specified condition in the Applicability with the LCO not met after performance of a risk assessment addressing inoperable systems and components, cannot be applied to this circumstance.

The OPERABILITY of two PORVs or an RCS vent opening of at least 2.9 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 325°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than 50°F above the RCS cold leg temperatures, or (2) the start of a charging/safety injection pump and its injection into a water-solid RCS.

The maximum allowed PORV setpoint for the Low Temperature Overpressure Protection System (LTOPS) is derived by analysis which models the performance

LOW TEMPERATURE OVERPRESSURE PROTECTION (Continued)

the Technical Requirements Manual.

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

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 noncondensible gases and/or steam from the Reactor Coolant System that could inhibit natural circulation core cooling. The OPERABILITY of 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.

3/4.7.8 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the Reactor Coolant System and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads.

Surveillance to demonstrate OPERABILITY is by performance of an augmented inservice inspection program specified in ~~the Technical Specification Equipment List Program, plant procedure PLP-106.~~ the Technical Requirements Manual. The program is in accordance with the ASME OM Code as required by 10 CFR 50.55a.