FENOC

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PY-CEI/NRR-2523L

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Perry Nuclear Power Plant
Docket No. 50-440
Operating License Amendment Request Pursuant to 10CFR50.90: Incorporation of Generic, Administrative Changes into the Perry Nuclear Power Plant Technical Specifications

Ladies and Gentlemen:

Nuclear Regulatory Commission (NRC) review and approval of a license amendment is requested for the Perry Nuclear Power Plant (PNPP). This request includes fourteen of the simpler, generic administrative/editorial/consistency improvements agreed upon between the Nuclear Energy Institute (NEI) Technical Specification Task Force (TSTF) and the NRC, subsequent to the conversion of the PNPP Technical Specifications to the improved Standard Technical Specifications.

Attachment 1 provides a summary of the overall package. Attachments 2 through 15 provide a description of the proposed changes for each TSTF item, a justification, and the annotated Technical Specification pages. Attachments 2 through 15 also provide annotated Bases pages for each TSTF item, for information, since the Bases are not a formal part of the Technical Specifications. Attachment 16 provides the Significant Hazards Consideration for the entire group of administrative changes.

The proposed changes are not required to support the next PNPP refueling outage, so they do not have a firm need date associated with them. These items have been grouped into one submittal since, due to their administrative/editorial/consistency nature, they can be enveloped by a generic Significant Hazards Consideration. It is recognized that they may be issued as one or as multiple amendments.

There are no regulatory commitments contained in this letter or its attachments. If you have questions or require additional information, please contact Mr. Gregory A. Dunn, Manager - Regulatory Affairs, at (440) 280-5305.

Very truly yours,

John K Wood

Attachments

A001

cc: NRC Project Manager
NRC Resident Inspector
NRC Region III
State of Ohio

I, John K. Wood, hereby affirm that (1) I am Vice President - Perry, of the FirstEnergy Nuclear Operating Company, (2) I am duly authorized to execute and file this certification as the duly authorized agent for The Cleveland Electric Illuminating Company, Toledo Edison Company, Ohio Edison Company, and Pennsylvania Power Company, and (3) the statements set forth herein are true and correct to the best of my knowledge, information and belief.

John K. Wood

John K. Wood

Subscribed to and affirmed before me, the 9th day of November, 2000

Sheryl A. Eckelberry

Sheryl A. Eckelberry
Notary Public
State of Ohio
My Commission Expires:
June 10, 2004

INTRODUCTION

The improved Standard Technical Specifications (iSTS) were adopted at the Perry Nuclear Power Plant (PNPP) in 1996, as an industry lead plant. Since that time, the Industry and the Nuclear Regulatory Commission (NRC) staff have worked to improve the new Standard specifications (NUREGs 1430 through 1434), and many generic changes have been developed. This process saves licensee and industry resources by resolving generic ideas once, rather than on each plant docket, and by pre-identifying the information necessary to process the change. This improves the adoption process for generically acceptable changes.

Generic changes to the iSTS NUREGs are proposed by the NEI Technical Specification Task Force (TSTF) to the NRC. The TSTF includes representatives from the four U.S. commercial nuclear power plant Owners Groups, and NEI. Generic changes are prepared and reviewed using a process that the TSTF and NRC developed to correct and improve the iSTS NUREGs. These proposed changes are assigned a number for tracking purposes, and are referred to as TSTF's (e.g., TSTF-2, TSTF-5, etc). After NRC approval, these TSTF's are available for adoption by plants.

This request includes fourteen of the simpler, generic administrative/editorial/consistency improvements agreed upon between the Nuclear Energy Institute (NEI) Technical Specification Task Force and the NRC since the PNPP conversion. It is expected that future requests will incorporate additional TSTF's.

The following Attachments provide PNPP-specific versions of the NRC-approved generic changes that are being requested at this time. For each of the requested changes, the following is provided:

- the associated TSTF number and it's short title,
- the specific changes requested to the PNPP Technical Specifications,
- a comparison between the requested change and the TSTF,
- the justification for the change (based upon the justification for the TSTF, with plant specific information added as needed), and
- identification of the specific affected Technical Specification pages.
- identification of the specific affected Bases pages, for information, since the Bases are not a formal part of the Technical Specifications.

Attachment 16 provides the Significant Hazards Consideration for the entire group of items.

The proposed changes are not required to support the next PNPP refueling outage, so they do not have a firm need date associated with them. These items have been grouped into one submittal since, due to their administrative/editorial/consistency nature, they can be enveloped by a single Significant Hazards Consideration. It is recognized that they may be issued as one or as multiple amendments.

A Table of Contents for the package is provided on the next page

TABLE OF CONTENTS

TS Page	Att. # / TSTF #
2.0-1	2 / 5
2.0-2	2 / 5
3.0-2	7 / 104; 12 / 166
3.1-7	3 / 32
3.1-8	3 / 32
3.4-21	11 / 153
3.4-24	11 / 153
3.8-25	4 / 38
3.8-32	14 / 278
3.8-33	14 / 278
3.9-10	11 / 153
3.9-13	11 / 153
5.0-2	6 / 65
5.0-3	6 / 65; 13 / 258
5.0-4	13 / 258
5.0-9	13 / 258
5.0-10	15 / 279
5.0-13	8 / 106; 9 / 118
5.0-15a	5 / 52
5.0-16	10 / 152
5.0-17	13 / 258
5.0-19	6 / 65

Bases Page	Att. # / TSTF #
B 2.0-4	2 / 5
B 2.0-5	2 / 5
B 2.0-6	2 / 5
B 2.0-8	2 / 5
B 2.0-9	2 / 5
B 2.0-10	2 / 5
B 3.0-6	7 / 104
B 3.0-12	5 / 52
B 3.1-15	3 / 32
B 3.1-16	3 / 32
B 3.1-17	3 / 32
B 3.4-45	11 / 153
B 3.4-50	11 / 153
B 3.8-56	4 / 38
B 3.9-26	11 / 153
B 3.9-31	11 / 153

TSTF 5 Delete notification, reporting, and restart requirements if a safety limit is violated

a. Requested Change

Retain the shutdown requirements if a safety limit (SL) is violated, as outlined in TS Section 2.2, SL Violations, but delete notification, reporting, and restart requirements from the Technical Specifications.

b. Consistency with TSTF

The proposed change is consistent with the TSTF.

c. Justification

This change deletes requirements from the Technical Specifications that are purely administrative or are adequately addressed by other regulatory controls:

- Safety Limit (SL) 2.2.1 requires the NRC Operations Center to be notified. This is addressed by 10CFR50.36(c)(1)(i)(A) ("The licensee shall notify the Commission as required by §50.72..."), and by 10CFR50.72 "Immediate Notification Requirements for Operating Nuclear Power Reactors".
- SL 2.2.3 requires "Within 24 hours, notify the plant manager and the corporate executive responsible for overall plant nuclear safety." Deleting this 24-hour reporting time frame is acceptable since it is only reasonable to expect utility management to be promptly informed of a Safety Limit violation considering their plant has been required to shut down, and restart of the plant must be authorized by the NRC. Therefore, post-change, this item's intent would still be met. Such a report to utility management also does not meet 10CFR50.36 criteria for retention in the Technical Specifications.
- SL 2.2.4 requires a 30-day Licensee Event Report (LER) to be submitted to the NRC, and to the plant manager and the corporate executive responsible for overall plant nuclear safety. Submittal of the report to the NRC is addressed by 10CFR50.36(c)(1)(i)(A) ("The licensee shall ... submit a Licensee Event Report to the Commission as required by §50.73."), and by 10CFR50.73 "Licensee Event Report System". "Submittal" to the plant manager and the corporate executive responsible for overall plant nuclear safety occurs as part of the LER review process, since they are part of the approval chain for the LER.
- SL 2.2.5 requires that "Operation of the unit shall not be resumed until authorized by the NRC." This is addressed by 10CFR50.36(c)(1)(i)(A) ("Operation must not be resumed until authorized by the Commission.")

d. Technical Specifications Affected

	TS Pages	Bases Pages
Safety Limit 2.2	2.0-1 2.0-2	B 2.0-4 B 2.0-5 B 2.0-6 B 2.0-8 B 2.0-9 B 2.0-10

Attachment 2
PY-CEI/NRR-2523L
Page 2 of 9

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 23.8% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.09 for two recirculation loop operation or ≥ 1.11 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

within 2 hours

With any SL violation, the following actions shall be completed:

2.2.1 Within 1 hour, notify the NRC Operations Center~~, in accordance with 10 CFR 50.72.~~ e

~~2.2.2 Within 2 hours:~~ e

2.2.2.1 Restore compliance with all SLs; and

2.2.2.2 Insert all insertable control rods.

2.2.3 Within 24 hours, notify the plant ~~manager and the corporate executive responsible~~ for overall plant nuclear safety. e

(continued)

All changes on this page made per Att. 2; TSTF 5

Attachment 2
PY-CEI/NRR-2523L
Page 3 of 9

2.2 SL Violations (continued)

2.2.4 Within 30 days, a Licensee Event Report (LER) shall be prepared pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC, the plant manager, and the corporate executive responsible for overall plant nuclear safety.

2.2.5 Operation of the unit shall not be resumed until authorized by the NRC.

All changes on this page made per Att. 2; TSTF 5

Attachment 2
PY-CEI/NRR-2523L
Page 4 of 9

BASES

APPLICABLE SAFETY ANALYSES (continued)

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2, the reactor vessel water level is required to be above the top of the active fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level becomes less than two thirds of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and also to provide adequate margin for effective action.

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel water level is greater than the top of the active irradiated fuel, in order to prevent elevated clad temperatures and resultant clad perforation.

APPLICABILITY

SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.

SAFETY LIMIT VIOLATIONS

~~2.2.1~~

~~If any SL is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 3).~~

2.2.~~2~~

Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. ~~4~~ 3). Therefore, it is required to insert all insertable control rods and restore compliance

(continued)

All changes on this page made per Att. 2; TSTF 5

Attachment 2
PY-CEI/NRR-2523L
Page 5 of 9

BASES

SAFETY LIMIT VIOLATIONS

2.2.2 (continued)

with the SL within 2 hours. These actions will include restoring reactor vessel water level in accordance with the Plant Emergency Instructions (e.g., manually initiating the ECCS or depressurizing the reactor vessel). The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

Per 10CFR50.36(c)(1)(i)(A), operation must not be resumed until authorized by the Nuclear Regulatory Commission.

~~2.2.3~~

~~If any SL is violated, the General Manager, Perry Nuclear Power Plant Department and the Vice President-Nuclear shall be notified within 24 hours. The 24 hour period provides time for plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management.~~

~~2.2.4~~

~~If any SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 5). A copy of the report shall also be submitted to the General Manager, Perry Nuclear Power Plant Department and the Vice President-Nuclear.~~

~~2.2.5~~

~~If any SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.~~

(continued)

All changes are consistent with Att. 2 and TSTF 5

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.

2. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel, GESTAR-II" (latest approved revision).

~~3. 10 CFR 50.72.~~ e

3A. 10 CFR 100.

~~5. 10 CFR 50.73.~~ e

All changes made per Att. 2; TSTF 5

Attachment 2
PY-CEI/NRR-2523L
Page 7 of 9

BASES

APPLICABLE SAFETY ANALYSES (continued)

The RCS pressure SL has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to ASME, Boiler and Pressure Vessel Code, Section III, 1971 Edition, including Addenda through the Winter of 1972 (Ref. 5), which permits a maximum pressure transient of 110%, 1375 psig, of design pressure 1250 psig. The SL of 1325 psig, as measured in the reactor steam dome, is equivalent to 1375 psig at the lowest elevation of the RCS. The RCS is currently designed to ASME Code, Section III, 1983 Edition, including addenda through the Winter of 1984 (Ref. 6), for the reactor recirculation piping, which permits a maximum pressure transient of 110% of design pressures of 1250 psig for suction piping, 1650 psig for discharge piping between the pump and the discharge valve, and 1550 psig beyond the discharge valve. The RCS pressure SL is selected to be the lowest transient overpressure allowed by the applicable codes.

SAFETY LIMITS

The maximum transient pressure allowable in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowable in the RCS piping, valves, and fittings is 110% of design pressures of 1250 psig for suction piping, 1650 psig for discharge piping between the pump and the discharge valve, and 1550 psig beyond the discharge valve. The most limiting of these allowances is the 110% of the suction piping design pressure; therefore, the SL on maximum allowable RCS pressure is established at 1325 psig as measured in the reactor steam dome.

APPLICABILITY

SL 2.1.2 applies in all MODES.

SAFETY LIMIT VIOLATIONS

<u>2.2.1</u>

If any SL is violated, the NRC ~~Operations Center~~ must be notified within ~~1 hour, in~~ accordance with 10 CFR 50.72 ~~(Ref. 7)~~.

(continued)

All changes made per Att. 2; TSTF 5

Attachment 2
PY-CEI/NRR-2523L
Page 8 of 9

BASES

SAFETY LIMIT VIOLATIONS (continued)

2.2.2

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. These actions will include restoring reactor vessel water level in accordance with the Plant Emergency Instructions (e.g., manually initiating the ECCS or depressurizing the reactor vessel). The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

~~2.2.3~~

~~If any SL is violated, the General Manager, Perry Nuclear Power Plant Department, and the Vice President-Nuclear shall be notified within 24 hours. The 24 hour period provides time for plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management.~~

~~2.2.4~~

~~If any SL is violated, a Licensee Event Report shall be prepared, reviewed and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 8). A copy of the report shall also be submitted to the General Manager, Perry Nuclear Power Plant Department, and the Vice President-Nuclear.~~

~~2.2.5~~

~~If any SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.~~

(continued)

All changes made per Att. 2; TSTF 5

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 14, and GDC 15.
2. ASME, Boiler and Pressure Vessel Code, Section III.
3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWA-5000.
4. 10 CFR 100.
5. ASME, Boiler and Pressure Vessel Code, 1971 Edition, Addenda, Winter of 1972.
6. ASME, Boiler and Pressure Vessel Code, 1983 Edition, Addenda, Winter of 1984.
~~7. 10 CFR 50.72.~~
~~8. 10 CFR 50.73.~~

All changes made per Att. 2; TSTF 5

TSTF 32 Slow/Stuck Control Rod Separation Criteria

a. Requested Change

Add a required Action to LCO 3.1.3, "Control Rod Operability", to confirm a control rod found to be stuck is properly separated from "slow" control rods.

b. Consistency with TSTF

The proposed change is consistent with the TSTF, except the Bases markup is modified to reflect the PNPP separation requirements.

c. Justification

This does not change any technical requirements for PNPP. For consistency reasons, it incorporates a "separation check" into LCO 3.1.3, "Control Rod Operability", like the one already required by LCO 3.1.4 "Control Rod Scram Times". The check helps to enforce the scram reactivity analyses. The analyses assume several slow control rods exist, in combination with a stuck rod, and a single-failure causes another control rod to fail to scram during the transient/ accident analysis. However, the analyses do not assume that the original stuck control rod is adjacent to any of the rods known to be "slow". If this condition occurs, the local scram reactivity rate assumed in the analysis might not be met. During the conversion to the iSTS, an extra requirement was included in PNPP LCO 3.1.4 (above & beyond the requirements in the iSTS NUREG) to ensure that when slow control rods are identified, the appropriate separation is verified between the known "slow" rods and any "withdrawn stuck" control rod. However, in LCO 3.1.3, which is where the operator is directed anytime a control rod is identified as stuck (rather than slow), a similar separation verification is not included. Therefore, a new Required Action A.1 is being added to LCO 3.1.3 to confirm that when a control rod is found to be "stuck", it is properly separated from control rods known to be "slow". The current Required Actions of Action A have been renumbered to reflect this addition. Corresponding changes to the Bases have also been marked.

This is an administrative change only, since it only makes LCO 3.1.3 consistent with LCO 3.1.4, and does not change the other requirements of LCO 3.1.3. It helps maintain the assumptions of the supporting safety analyses, thereby ensuring the intent of the Technical Specifications (to enforce the safety analyses) is met.

d. Technical Specifications Affected

	TS Pages	Bases Pages
LCO 3.1.3	3.1-7 3.1-8	B 3.1-15 B 3.1-16 B 3.1-17

Attachment 3
PY-CEI/NRR-2523L
Page 2 of 9

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Control Rod OPERABILITY

LCO 3.1.3 Each control rod shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each control rod.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One withdrawn control rod stuck.	------------NOTE------------ A stuck rod may be bypassed in the Rod Action Control System (RACS) in accordance with SR 3.3.2.1.9 if required to allow continued operation. ----------------------------	
	A.~~1~~2 Disarm the associated control rod drive (CRD). AND	2 hours
		(continued)

A.1 Verify stuck control rod separation criteria are met. Immediately

AND

All changes made per Att. 3; TSTF 32

Attachment 3
PY-CEI/NRR-2523L
Page 3 of 9

ACTIONS

CONDITION		REQUIRED ACTION	COMPLETION TIME
A. (continued) All changes per Att. 3; TSTF 32	A.~~2~~3	Perform SR 3.1.3.2 and SR 3.1.3.3 for each withdrawn OPERABLE control rod.	24 hours from discovery of Condition A concurrent with THERMAL POWER greater than or equal to the low power setpoint (LPSP) of the Rod Pattern Control System (RPCS).
	<u>AND</u>		
	A.~~3~~4	Perform SR 3.1.1.1.	72 hours
B. Two or more withdrawn control rods stuck.	B.1	Be in MODE 3.	12 hours
C. One or more control rods inoperable for reasons other than Condition A or B.	C.1	--------NOTE--------- Inoperable control rods may be bypassed in RACS in accordance with SR 3.3.2.1.9, if required, to allow insertion of inoperable control rod and continued operation. --------------------- Fully insert inoperable control rod.	3 hours
	<u>AND</u>		
	C.2	Disarm the associated CRD.	4 hours

(continued)

Attachment 3
PY-CEI/NRR-2523L
Page 4 of 9

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Control Rod Scram Times

LCO 3.1.4

a. No more than 13 OPERABLE control rods shall be "slow," in accordance with Table 3.1.4-1; and

b. No OPERABLE control rod that is "slow" shall occupy a location adjacent to another OPERABLE control rod that is "slow" or a withdrawn control rod that is stuck.

FYI

APPLICABILITY: MODES 1 and 2.

No changes to this page; for information for Att. 3; TSTF 32

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

-----------------------------------NOTE-----------------------------------
During single control rod scram time Surveillances, the control rod drive (CRD) pumps shall be isolated from the associated scram accumulator.

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify each control rod scram time is within the limits of Table 3.1.4-1 with reactor steam dome pressure ≥ 950 psig.	Prior to exceeding 40% RTP after fuel movement within the reactor pressure vessel AND (continued)

Table 3.1.4-1
Control Rod Scram Times

NOTES

1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

2. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to notch position 13. These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."

FYI

NOTCH POSITION	SCRAM TIMES(a)(b) (seconds) REACTOR STEAM DOME PRESSURE(c) 950 psig	REACTOR STEAM DOME PRESSURE(c) 1050 psig
43	0.30	0.31
29	0.78	0.84
13	1.40	1.53

(a) Maximum scram time from fully withdrawn position, based on de-energization of scram pilot valve solenoids as time zero.

(b) Scram times as a function of reactor steam dome pressure when < 950 psig are within established limits.

(c) For intermediate reactor steam dome pressures, the scram time criteria are determined by linear interpolation.

No changes to this page; for information for Att. 3; TSTF 32

BASES

LCO (continued)

satisfy the intended reactivity control requirements, strict control over the number and distribution of inoperable control rods is required to satisfy the assumptions of the DBA and transient analyses.

APPLICABILITY

In MODES 1 and 2, the control rods are assumed to function during a DBA or transient and are therefore required to be OPERABLE in these MODES. In MODES 3 and 4, control rods are not able to be withdrawn since the reactor mode switch is in Shutdown and a control rod block is applied. This provides adequate requirements for control rod OPERABILITY during these conditions. Control rod requirements in MODE 5 are located in LCO 3.9.5, "Control Rod OPERABILITY—Refueling."

ACTIONS

The ACTIONS table is modified by a Note indicating that a separate Condition entry is allowed for each control rod. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable control rod. Complying with the Required Actions may allow for continued operation, and subsequent inoperable control rods are governed by subsequent Condition entry and application of associated Required Actions.

A.1, A.2, ~~and~~ A.3, and A.4

A control rod is considered stuck if it will not insert (using all available insertion methods) by either CRD drive water or scram pressure. With a fully inserted control rod stuck, no actions are required as long as the control rod remains fully inserted. The Required Actions are modified by a Note that allows a stuck control rod to be bypassed in the Rod Action Control System (RACS) to allow continued operation. SR 3.3.2.1.9 provides additional requirements when control rods are bypassed in RACS to ensure compliance with the CRDA analysis. With one withdrawn control rod stuck, Insert the control rod must be disarmed within 2 hours. The allowed Completion Time of 2 hours is acceptable, considering the reactor can still be shut down, assuming no additional control rods fail to insert, and provides a reasonable amount of time to perform the Required Action in an orderly manner. Isolating the control rod from scram prevents damage to the CRDM. The control rod can be

(continued)

All changes per Att. 3; TSTF 32

Insert for Bases Page B 3.1-15

...the local scram reactivity rate assumptions may not be met if the stuck control rod separation criteria are not met. Therefore, verification that the separation criteria are met must be performed immediately. The stuck control rod separation criteria are that the stuck control rod may not occupy a location adjacent to a "slow" control rod. The description of "slow" control rods is provided in LCO 3.1.4 "Control Rod Scram Times". In addition, ...

All changes made per Att. 3;
TSTF 32

Attachment 3
PY-CEI/NRR-2523L
Page 8 of 9

BASES

All changes made per Att. 3; TSTF 32

ACTIONS

A.1, A.2, ~~and~~ A.3 and A.4 (continued)

isolated from scram by isolating the hydraulic control unit from scram and normal drive and withdraw pressure, yet still maintain cooling water to the CRD. A control rod can be hydraulically disarmed by closing the drive water and exhaust water isolation valves. Electrically, the control rod can be disarmed by disconnecting power from all four directional control valve solenoids.

Monitoring of the insertion capability for each withdrawn control rod must also be performed within 24 hours. SR 3.1.3.2 and SR 3.1.3.3 perform periodic tests of the control rod insertion capability of withdrawn control rods. Testing each withdrawn control rod ensures that a generic problem does not exist. The allowed Completion Time of 24 hours provides a reasonable time to test the control rods, considering the potential for a need to reduce power to perform the tests. Required Action A.2 has a modified time zero Completion Time. The 24 hour Completion Time for this Required Action starts when the withdrawn control rod is discovered to be stuck and THERMAL POWER is greater than the actual low power setpoint (LPSP) of the rod pattern controller (RPC), since the notch insertions may not be compatible with the requirements of rod pattern control (LCO 3.1.6) and the RPC (LCO 3.3.2.1, "Control Rod Block Instrumentation").

To allow continued operation with a withdrawn control rod stuck, an evaluation of adequate SDM is also required within 72 hours. Should a DBA or transient require a shutdown, to preserve the single failure criterion an additional control rod would have to be assumed to have failed to insert when required. Therefore, the original SDM demonstration may not be valid. The SDM must therefore be evaluated (by measurement or analysis) with the stuck control rod at its stuck position and the highest worth OPERABLE control rod assumed to be fully withdrawn.

The allowed Completion Time of 72 hours to verify SDM is adequate, considering that with a single control rod stuck in a withdrawn position, the remaining OPERABLE control rods are capable of providing the required scram and shutdown reactivity. Failure to reach MODE 4 is only likely if an additional control rod adjacent to the stuck control rod

(continued)

All changes per Att. 3; TSTF 32

ACTIONS

A.1, A.2, ~~and~~ A.3, and A.4 (continued)

also fails to insert during a required scram. Even with the postulated additional single failure of an adjacent control rod to insert, sufficient reactivity control remains to reach and maintain MODE 3 conditions (Ref. 7).

B.1

With two or more withdrawn control rods stuck, the plant should be brought to MODE 3 within 12 hours. Isolating the control rod from scram prevents damage to the CRDM. The occurrence of more than one control rod stuck at a withdrawn position increases the probability that the reactor cannot be shut down if required. Insertion of all insertable control rods eliminates the possibility of an additional failure of a control rod to insert. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

With one or more control rods inoperable for reasons other than being stuck in the withdrawn position, operation may continue, provided the control rods are fully inserted within 3 hours and disarmed (electrically or hydraulically) within 4 hours. Inserting a control rod ensures the shutdown and scram capabilities are not adversely affected. The control rod is disarmed to prevent inadvertent withdrawal during subsequent operations. The control rods can be hydraulically disarmed by closing the drive water and exhaust water isolation valves. Electrically, the control rods can be disarmed by disconnecting power from all four directional control valve solenoids. With a control rod not coupled to its associated drive mechanism, insert the control rod drive mechanism to accomplish recoupling. Verify recoupling by withdrawing the control rod and observing any indicated response of the nuclear instrumentation and demonstrating that the control rod drive will not go to the overtravel position. Required Action C.1 is modified by a Note that allows control rods to be bypassed in the RACS if required to allow insertion of the inoperable control rods and continued operation. SR 3.3.2.1.9 provides additional requirements when the control rods are bypassed to ensure compliance with the CRDA analysis.

(continued)

TSTF 38 Revise visual surveillance of batteries to specify inspection is for performance degradation

a. Requested Change

Clarify the requirements of SR 3.8.4.3, battery visual inspection, to be consistent with the original intent, and with the present wording of the Bases. The required inspection is for physical damage or abnormal deterioration which could potentially degrade battery performance.

b. Consistency with TSTF

The proposed change is consistent with the TSTF.

c. Justification

The Bases of SR 3.8.4.3 in NUREG-1434 state that this SR "provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance." As a result, it is interpreted that physical damage or abnormal deterioration has to be of a type that could degrade battery performance before the SR would fail to be met. The presence of physical damage or deterioration does not necessarily represent a failure of SR 3.8.4.3, provided an evaluation determines that the physical damage or deterioration does not affect the OPERABILITY of the battery (its ability to perform its design function). Therefore, for consistency with the Bases for SR 3.8.4.3 in NUREG-1434, SR 3.8.4.3 is being revised to add the words "that could degrade battery performance." The Bases for SR 3.8.4.3 are also being revised to clarify measures to be taken in the event physical damage or abnormal deterioration are discovered.

This is an administrative change only, since it only makes the SR consistent with the Bases, and does not change the original intent of the Surveillance Requirement.

d. Technical Specifications Affected

	TS Pages	Bases Pages
SR 3.8.4.3	3.8-25	B 3.8-56

Attachment 4
PY-CEI/NRR-2523L
Page 2 of 3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.4.1	Verify battery terminal voltage is ≥ 129 V on float charge.	7 days
SR 3.8.4.2	Verify no visible corrosion at battery terminals and connectors. <u>OR</u> Verify battery connection resistance is ≤ 5.0 E-5 ohm for inter-cell connections, ≤ 5.0 E-5 ohm for inter-rack connections, ≤ 5.0 E-5 ohm for inter-tier connections, ≤ 5.0 E-5 ohm for terminal connections; for Div 1 and Div 2 and ≤ 1.0 E-4 ohm for inter-cell connections, ≤ 1.0 E-4 ohm for inter-rack connections, ≤ 1.0 E-4 ohm for inter-tier connections, ≤ 1.0 E-4 ohm for terminal connections. for Div 3.	92 days
SR 3.8.4.3	Verify battery cells, cell plates, and racks show no visual indication of physical damage or abnormal deterioration ~~.~~ that could degrade battery performance	18 months
SR 3.8.4.4	Remove visible corrosion, and verify battery cell to cell and terminal connections are coated with anti-corrosion material.	18 months

(continued)

All changes per Att. 4;
TSTF 38

Attachment 4
PY-CEI/NRR-2523L
Page 3 of 3

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.4.3

Visual inspection of the battery cells, cell plates, and battery racks provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance.

The presence of physical damage or deterioration does not necessarily represent a failure of this SR, provided an evaluation determines that the physical damage or deterioration does not affect the OPERABILITY of the battery (its ability to perform its design function).

The 18 month Frequency of the Surveillance is based on engineering judgement, taking into consideration the desired unit conditions to perform the Surveillance. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.8.4.4 and SR 3.8.4.5

Visual inspection and resistance measurements of inter-cell, inter-rack, inter-tier, and terminal connections provides an indication of physical damage or abnormal deterioration that could indicate degraded battery condition. The anti-corrosion material is used to ensure good electrical connections and to reduce terminal deterioration. The visual inspection for corrosion is not intended to require removal of and inspection under each terminal connection.

The removal of visible corrosion is a preventive maintenance SR. The presence of visible corrosion does not necessarily represent a failure of this SR, provided visible corrosion is removed during performance of this Surveillance.

The 18 month Frequency of the Surveillance is based on engineering judgement, taking into consideration the desired unit conditions to perform the Surveillance. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

All changes per Att. 4, TSTF 38

TSTF 52 Implement 10 CFR 50, Appendix J, Option B

a. Requested Change

The PNPP Technical Specifications were already revised to implement 10CFR50 Appendix J, Option B; however, the PNPP changes were made before this TSTF was approved. In finalizing the wording of this TSTF, one item was incorporated that is not reflected in the PNPP Technical Specifications. This item deals with a sentence included in the Section 5 Program entitled "Primary Containment Leakage Rate Testing Program". The sentence states "The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program." [Informational Note: SR 3.0.2 states "The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency,..."]

The NRC and the TSTF extensively debated whether the Section 3.0 provisions (such as SR 3.0.2) generically apply to the Section 5 Administrative Controls. The final decision was that Section 3.0 does not generically apply to Section 5. In the iSTS NUREG, there are no instances in which Section 5 programs state that Section 3.0 provisions do **not** apply, only that they **do** apply. To avoid confusion by stating in one location that SR 3.0.2 does not apply (thereby implying that it does generically apply in other locations in Section 5), the statement is being replaced. The replacement sentence states "Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J". A corresponding change is also made in the Bases for SR 3.0.2.

b. Consistency with TSTF

The proposed change is consistent with the TSTF for this one item. The PNPP Technical Specifications are already consistent with other portions of the TSTF.

c. Justification

This change will preclude the possibility of a licensee misconstruing the SR 3.0.2 allowance and applying it to either the 10 CFR 50, Appendix J testing frequencies or other frequencies in Technical Specification Section 5 for which it is not specifically permitted. Other changes similar to this are being proposed for TSTFs 118 and 258 (Attachments 9 & 13).

This is an administrative change only, since it does not change the original intent of the Specification, i.e., to not permit additional extensions of testing frequencies specified by 10 CFR 50, Appendix J.

d. Technical Specifications Affected

	TS Pages	Bases Pages
Administrative Control 5.5.12	5.0-15a	B 3.0-12

Attachment 5
PY-CEI/NRR-2523L
Page 2 of 4

5.5.7 Ventilation Filter Testing Program (VFTP) (continued)

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980 at the system flowrate specified below ± 10%:

ESF Ventilation System	Delta P	Flowrate
a) Control Room Emergency Recirculation	4.9" H_2O	30,000 scfm
b) Fuel Handling Building	4.9" H_2O	15,000 scfm
c) Annulus Exhaust Gas Treatment	6.0" H_2O	2,000 scfm

e. Demonstrate that the heaters for each of the ESF systems dissipate the value specified below ± 10% when corrected to nominal input voltage when tested in accordance with ANSI N510-1980:

ESF Ventilation System	Wattage
a) Control Room Emergency Recirculation	100 kW
b) Fuel Handling Building	50 kW
c) Annulus Exhaust Gas Treatment	20 kW

FYI → The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.8 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the main condenser offgas treatment system, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

The program shall include:

a. The limits for concentrations of hydrogen in the main condenser offgas treatment system and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion); and

(continued)

Attachment 5
PY-CEI/NRR-2523L
Page 3 of 4

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

- BN-TOP-1 methodology may be used for Type A tests.
- The corrections to NEI 94-01 which are identified on the Errata Sheet attached to the NEI letter, "Appendix J Workshop Questions and Answers," dated March 19, 1996 are considered an integral part of NEI 94-01.
- The containment isolation check valves in the Feedwater penetrations are tested per the Inservice Testing Program (Technical Specification 5.5.6).

The peak calculated primary containment internal pressure for the design basis loss of coolant accident is 6.40 psig. For conservatism P_a is defined as 7.80 psig.

The maximum allowable primary containment leakage rate, L_a, shall be 0.20% of primary containment air weight per day at the peak containment pressure (P_a).

Leakage rate acceptance criteria are:

a. Primary containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. However, during the first unit startup following testing performed in accordance with this Program, the leakage rate acceptance criteria are $< 0.6\ L_a$ for the Type B and Type C tests, and $\leq 0.75\ L_a$ for the Type A tests;

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is ≤ 2.5 scfh when tested at $\geq P_a$.

   2) For each door, leakage rate is ≤ 2.5 scfh when the gap between the door seals is pressurized to $\geq P_a$.

~~The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.~~

The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

Nothing in these Technical Specifications shall be construed to modify the testing Frequencies required by 10 CFR 50, Appendix J.

(continued)

All changes per Att. 5; TSTF 52

Attachment 5
PY-CEI/NRR-2523L
Page 4 of 4

SR 3.0.2 (continued)

All changes per Att. 5; TSTF 52

The 25% extension does not significantly degrade the reliability that results from performing the Surveillance at its specified Frequency. This is based on the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the SRs. The exceptions to SR 3.0.2 are those Surveillances for which the 25% extension of the interval specified in the Frequency does not apply. These exceptions are stated in the individual Specifications. The requirements of regulations take precedence over the TS. ~~Therefore, when a test interval is specified in the regulations, the test interval cannot be extended by the TS, and the Surveillance Requirements will then include a NOTE in the frequency stating "SR 3.0.2 is not applicable." The statement "SR 3.0.2 is not applicable" can also be used in cases where the test interval is not specified in the regulations. An example is the statement in the Primary Containment Leakage Rate Testing Program that "the provisions of SR 3.0.2 do not apply. . ." This exception is provided because the Program already includes extension of test intervals.~~

An example is in the Primary Containment Leakage Rate Testing Program. This program establishes testing requirements and Frequencies in accordance with the requirements of regulations. The TS cannot in and of themselves extend a test interval specified in the regulations.

As stated in SR 3.0.2, the 25% extension also does not apply to the initial portion of a periodic Completion Time that requires performance on a "once per..." basis. The 25% extension applies to each performance after the initial performance. The initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner.

The provisions of SR 3.0.2 are not intended to be used repeatedly merely as an operational convenience to extend Surveillance intervals (other than those consistent with refueling intervals) or periodic Completion Time intervals beyond those specified.

SR 3.0.3

SR 3.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified Frequency. A delay period of up to 24 hours or up to the limit of the specified

(continued)

TSTF 65 Use of generic titles for utility positions

a. Requested Change

The PNPP Technical Specifications already reflect generic titles in most instances; however, it was done before this TSTF was approved. In finalizing the wording of this TSTF, one item was incorporated that is not fully reflected in the PNPP Technical Specifications. This item is the use of the term "radiation protection" in lieu of "health physics". This change is proposed for inclusion on pages 5.0-2, 5.0-3 and 5.0-19.

b. Consistency with TSTF

The proposed change is consistent with the TSTF for this one item. The PNPP Technical Specifications are already consistent with other portions of the TSTF. Note: The TSTF changes the term "health physics" to "radiation protection" in two places (once in Section 5.2.2 "Unit Staff", and once in Section 5.7.1 "High Radiation Area"). The term "health physics" is used in several other places in Section 5, so these other locations have also been revised for consistency.

c. Justification

At PNPP, the term "radiation protection" has been adopted in lieu of "health physics". Therefore, references to health physics should be revised to be consistent with plant terminology. Changing this title has no effect on plant safety, and it has no impact on the effectiveness of administrative controls over the health physics/radiation protection function. Also, the NRC recently (Amendment 111) approved a similar change for PNPP, however, these additional locations of the term "health physics" were not addressed in that revision.

This is an administrative change only, since it only changes the generic term used to refer to health physics/radiation protection personnel, it provides consistency throughout the Technical Specifications, and it does not change the intent of the function performed by these personnel.

d. Technical Specifications Affected

	TS Pages	**Bases Pages**
Administrative Control 5.2.1	5.0-2	N/A
Administrative Control 5.2.2	5.0-3	
Administrative Control 5.7.1	5.0-19	

Attachment 6
PY-CEI/NRR-2523L
Page 2 of 5

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the plant specific titles of the personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the USAR;

b. The plant manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant;

c. A specified corporate executive shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety; and

d. The individuals who train the operating staff, carry out ~~health physics~~, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

(continued)

radiation protection duties

Change is per Att. 6;
& is consistent with TSTF 65

5.2.2 Unit Staff

The unit staff organization shall include the following:

a. A non-licensed operator shall be on site when fuel is in the reactor vessel, and an additional non-licensed operator shall be on site while the unit is in MODE 1, 2, or 3.

This change per Att. 13; TSTF 258

b. Deleted ~~At least one licensed Reactor Operator (RO) shall be present in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, or 3, at least one licensed Senior Reactor Operator (SRO) shall be present in the control room.~~

c. Shift crew composition may be one less than the minimum requirements of 10 CFR 50.54(m)(2)(i) and Specifications 5.2.2.a and 5.2.2.g for a period of time not to exceed two hours in order to accommodate unexpected absence of on-duty shift crew members, provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

These changes per Att. 6; & consistent with TSTF 65

d. A radiation protection ~~health physics~~ technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

e. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety related functions (e.g., licensed SROs, licensed ROs, radiation protection technicians, ~~health physicists~~, auxiliary operators, and key maintenance personnel). The procedures shall include guidelines on working hours that ensure that adequate shift coverage is maintained without routine heavy use of overtime.

Any deviation from the working hour guidelines shall be authorized in advance by the Plant Manager or his designees, in accordance with approved administrative procedures, or by higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation.

Controls shall be included in the procedures such that the individual overtime shall be reviewed monthly by the Plant Manager or his designees to ensure that excessive hours have not been assigned. Routine deviation from the working hour guidelines is not authorized.

(continued)

Attachment 6
PY-CEI/NRR-2523L
Page 4 of 5

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

5.7.1 Pursuant to 10 CFR 20, paragraph 20.1601(c), in lieu of the requirements of 10 CFR 20.1601(a), each high radiation area, as defined in 10 CFR 20, in which the intensity of radiation is > 100 mrem/hr but < 1000 mrem/hr, shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., radiation protection technicians) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

FYI

FYI

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.

FYI

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the ~~health physics~~ radiation protection supervisor in the RWP.

5.7.2 In addition to the requirements of Specification 5.7.1, areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose ≥ 1000 mrem shall be provided with locked or continuously guarded doors to prevent unauthorized entry and the keys shall be maintained under the administrative control of the shift supervisor on duty or the radiation protection supervisor. Doors shall remain locked except during periods of access by personnel under an approved RWP.

FYI

All changes per Att. 6; TSTF 65

(continued)

5.7 High Radiation Area

5.7.2 (continued) No changes to this page; for information only

FYI → Individuals qualified in radiation protection procedures (e.g., radiation protection technicians) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates ≤ 3000 mrem/hr, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas.

5.7.3 In addition to the requirements of Specification 5.7.1, for individual high radiation areas accessible to personnel with radiation levels such that a major portion of the body could receive in 1 hour a dose ≥ 1000 mrem that are located within large areas such as reactor containment, where no enclosure exists for purposes of locking, or that are not continuously guarded, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded and conspicuously posted, and a flashing light shall be activated as a warning device.

5.7.4 In addition to the requirements and exemptions of Specifications 5.7.1 and 5.7.2 for individual areas accessible to personnel such that a major portion of the body could receive in 1 hour a dose > 3000 mrem, entry shall require an approved RWP which will specify dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, continuous surveillance, direct or remote, such as use of closed circuit TV cameras, may be made by personnel qualified in radiation protection procedures to provide positive exposure control over activities within the areas.

TSTF 104 Relocate to the Bases the discussion of Exceptions to LCO 3.0.4

a. Requested Change

This change removes the additional discussion provided in LCO 3.0.4 with respect to the use of exceptions, and provides the necessary discussion in the Bases.

b. Consistency with TSTF

The proposed change is consistent with the TSTF.

c. Justification

This change provides consistency with LCO 3.0.3 by moving to the Bases the discussion of what an exception to the LCO allows. In addition, this change reduces the potential for confusion by revising the new Bases discussion to eliminate the repeated use of the phrase "MODES or other specified conditions in the Applicability" to increase clarity.

This change is administrative only since it provides consistency between LCOs 3.0.3 and 3.0.4, and does not change the intent of exceptions to LCO 3.0.4. It simply relocates a sentence discussing what an "exception" to LCO 3.0.4 allows.

d. Technical Specifications Affected

	TS Pages	Bases Pages
LCO 3.0.4	3.0-2	B 3.0-6

Attachment 7
PY-CEI/NRR-2523L
Page 2 of 3

3.0 LCO APPLICABILITY

LCO 3.0.4 (continued)

specified conditions in the Applicability that are required to comply with ACTIONS, or that are part of a shutdown of the unit.

Exceptions to this Specification are stated in the individual Specifications. These exceptions allow ~~entry~~ into MODES or other specified conditions ~~in the~~ Applicability when the associ~~ated~~ ACTIONS to be entered allow unit operati~~on in the~~ MODE or other specified conditi~~on in the~~ Applicability only for a limited period of ~~time~~.

This change per Att. 7; TSTF 104

LCO 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, and 3.

LCO 3.0.5

Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

LCO 3.0.6

When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an ~~additional~~ evaluation ~~and limitations may be required~~ shall be performed in accordance with Specification 5.5.10, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

This change per Att. 12; TSTF 166

When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

(continued)

Attachment 7
PY-CEI/NRR-2523L
Page 3 of 3

BASES

LCO 3.0.4 (continued)

provisions of LCO 3.0.4 shall not prevent changes in MODES or other specified conditions in the Applicability that result from any unit shutdown.

Exceptions to LCO 3.0.4 are stated in the individual Specifications. Exceptions may apply to all the ACTIONS or to a specific Required Action of a Specification.

The exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered do not provide for continued operation for an unlimited period of time.

Surveillances do not have to be performed on the associated inoperable equipment (or on variables outside the specified limits), as permitted by SR 3.0.1. Therefore, changing MODES or other specified conditions while in an ACTIONS Condition, either in compliance with LCO 3.0.4, or where an exception to LCO 3.0.4 is stated, is not a violation of SR 3.0.1 or SR 3.0.4 for those Surveillances that do not have to be performed due to the associated inoperable equipment. However, SRs must be met to ensure OPERABILITY prior to declaring the associated equipment OPERABLE (or variable within limits) and restoring compliance with the affected LCO.

LCO 3.0.4 is only applicable when entering MODE 3 from MODE 4, MODE 2 from MODE 3 or 4, or MODE 1 from MODE 2. Furthermore, LCO 3.0.4 is applicable when entering any other specified condition in the Applicability only while operating in MODE 1, 2, or 3. The requirements of LCO 3.0.4 do not apply in MODES 4 and 5, or in the other specified conditions of the Applicability (unless in MODE 1, 2, or 3) because the ACTIONS of individual Specifications sufficiently define the remedial measures to be taken.

LCO 3.0.5

LCO 3.0.5 establishes the allowance for restoring equipment to service under administrative controls when it has been removed from service or declared inoperable to comply with ACTIONS. The sole purpose of this Specification is to provide an exception to LCO 3.0.2 (e.g., to not comply with the applicable Required Action(s)) to allow the performance of SRs to demonstrate:

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

(continued)

All changes on this page made per Att. 7; TSTF 104

TSTF 106 Change to Diesel Fuel Oil Testing Program

a. Requested Change

This change will make it clear that the Technical Specification requirement is only applicable to the new fuel, must be done within 31 days following addition of fuel to the storage tanks, and is only required to be done one time.

b. Consistency with TSTF

The proposed change is consistent with the TSTF.

c. Justification

As currently worded, paragraph b of the Diesel Fuel Oil Testing Program can be, and has been, misinterpreted. This change is administrative only, since it does not change the intent of the current SR, it simply clarifies the original intent.

d. Technical Specifications Affected

	TS Page	Bases Page
Administrative Control 5.5.9	5.0-13	N/A

Attachment 8
PY-CEI/NRR-2523L
Page 2 of 2

5.5.8 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

b. A surveillance program to ensure that the quantity of radioactivity contained in any temporary outdoor tanks not including liners for shipping radwaste is ≤ 10 curies, excluding tritium and dissolved or entrained noble gases.

FYI

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.9 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:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

1. an API gravity or an absolute specific gravity within limits,

2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

3. a clear and bright appearance with proper color;

, verify that the properties of the new fuel oil, other than those addressed in a, above, are within limits for ASTM 2D fuel oil

b. ~~Other properties of the new fuel oil are within limits for ASTM 2D fuel oil~~ Within 31 days ~~of~~ following addition of the new fuel oil to storage tanks~~;~~ and

c. Total particulate concentration of the fuel oil in the storage tanks is ≤ 10 mg/l when tested every 31 days in accordance with ASTM D-2276.

Insert new ¶

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program testing frequencies.

(continued)

Changes to item b. are per Att. 8 ; TSTF 106
The new paragraph is per Att. 9 ; TSTF 118

TSTF 118 Administrative Controls Program Exceptions

a. Requested Change

Administrative Control 5.5.9 "Diesel Fuel Oil Testing Program" is revised to add the following sentence, "The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program testing frequencies."

b. Consistency with TSTF

The proposed change is consistent with the TSTF.

c. Justification

The NRC and the TSTF extensively debated whether the Section 3.0 provisions (such as SR 3.0.2) generically apply to the Section 5 Administrative Controls. The NRC position was that Section 3.0 does apply to Chapter 5. In a meeting on 12/15/98, and in a letter dated June 28, 1999 from W. Beckner (NRC) to J. Davis (NEI), the staff stated "It is the staff position that SR 3.0.2 does apply to the frequencies that are explicitly stated in Section 5.0 (i.e., Diesel Fuel Oil Testing Program)...". However, the final decision was that Section 3.0 does not generically apply to Chapter 5, since in the iSTS NUREG, there are no instances in which Section 5 programs state that Section 3.0 provisions do **not** apply, only that they **do** apply. Based on this final agreement, Administrative Control 5.5.9 is revised to add the following sentence, "The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program testing frequencies." This sentence provides consistency with the current application of these requirements as provided in other programs (e.g., 5.5.6 "Inservice Testing Program", 5.5.7 "Ventilation Filter Testing Program", and 5.5.8 "Explosive Gas and Storage Tank Radioactivity Monitoring Program"). SR 3.0.2 and SR 3.0.3 are already applicable to the Surveillance Requirements which reference these programs, and, therefore, the lack of an applicability statement in this program introduces confusion. Further, the applicability of SR 3.0.2 and SR 3.0.3 to the program surveillances is consistent with the pre-iSTS conversion licensing basis and the old STS, since SR 3.0.2 and SR 3.0.3 previously applied to these fuel oil SRs. Other changes similar to this are made in TSTFs 52 and 258.

This change is an administrative clarification only, since it provides consistency with the requirements for other programs in Section 5.0, and it maintains the original intent of the NRC by applying SR 3.0.2 and SR 3.0.3 to this program.

d. Technical Specifications Affected

	TS Page	Base Pages
Administrative Control 5.5.9	5.0-13	N/A

Attachment 9
PY-CEI/NRR-2523L
Page 2 of 2

5.5.8 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

b. A surveillance program to ensure that the quantity of radioactivity contained in any temporary outdoor tanks not including liners for shipping radwaste is ≤ 10 curies, excluding tritium and dissolved or entrained noble gases.

FYI

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.9 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:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

1. an API gravity or an absolute specific gravity within limits,

2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

3. a clear and bright appearance with proper color;

, verify that the properties of the new fuel oil, other than those addressed in a., above, are within limits for ASTM 2D fuel oil

b. ~~Other properties of the new fuel oil are within limits for ASTM 2D fuel oil~~ Within 31 days ~~of~~ following addition of the new fuel oil to storage tanks~~;~~ and

c. Total particulate concentration of the fuel oil in the storage tanks is ≤ 10 mg/l when tested every 31 days in accordance with ASTM D-2276.

Insert new ¶

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program testing frequencies.

(continued)

Changes to item b. are per Att. 8 ; TSTF 106
The new paragraph is per Att. 9 ; TSTF 118

TSTF 152 Revise Reporting Requirements to be Consistent with 10 CFR 20.

a. Requested Change

This change revises Section 5.6.1 "Occupational Radiation Exposure Report", to be consistent with a letter by C. I. Grimes (NRC) dated 7/28/95, on changes to Technical Specifications resulting from revisions to 10CFR20.

b. Consistency with TSTF

The proposed change is consistent with the TSTF with the following clarification:

- The word "calendar" is excluded from the last sentence of the TSTF Section 5.6.1 insert, regarding the submittal date for the Occupational Radiation Exposure Report, due to changes already made to this sentence by Amendment 111.
- The PNPP Technical Specifications already reflect the important changes proposed by the TSTF to Section 5.6.3 "Radioactive Effluent Release Report", so no changes are proposed to this section.

c. Justification

The NRC provided guidance on how to change the Technical Specifications in a proposed Generic Letter that was never issued. Instead, this TSTF was processed to provide generic industry guidance on how to revise the Technical Specifications. These changes are administrative, since they reflect the rule changes.

d. Technical Specifications Affected

	TS Pages	**Bases Pages**
Administrative Control 5.6.1	5.0-16	N/A

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

Attachment 10
PY-CEI/NRR-2523L
Page 2 of 2

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Occupational Radiation Exposure Report

A tabulation on an annual basis of the number of ~~station~~, utility, and other personnel (including contractors) for whom monitoring was ~~required~~ performed, receiving ~~exposures~~ an annual deep dose equivalent > 100 ~~mrem/yr~~ mrem and the associated ~~man rem exposure~~ collective deep dose equivalent (reported in person-rem) according to work and job functions, (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket ~~dosimeter~~ ionization chamber, thermoluminescent dosimeter (TLD), electronic dosimeter, or film badge measurements. Small exposures totalling < 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total ~~whole body dose~~ deep dose equivalent received from external sources should be assigned to specific major work functions.

The Occupational Radiation Exposure Report covering the activities of the unit for the previous year shall be submitted by April 30 of each year.

5.6.2 Annual Radiological Environmental Operating Report

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

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated

(continued)

All changes per Att. 10; TSTF 152

TSTF 153 Clarify Exception Notes to be Consistent with the Requirement Being Excepted

a. Requested Change

Several LCOs have Notes that contain exceptions to the LCOs and have wording that is inconsistent with the wording of LCO. The Notes are revised to provide consistent wording with the requirement being excepted. LCOs 3.4.9, 3.4.10, 3.9.8, and 3.9.9 are revised to correct inconsistent wording.

b. Consistency with TSTF

The proposed change is consistent with the TSTF with the following clarification:

- PNPP LCO 3.4.10 has a Note 3 that is not in the BWR/6 iSTS NUREG. This Note has the same wording problem that is corrected by this TSTF. The wording of this LCO Note 3 is also corrected by this proposed change.

c. Justification

The Residual Heat Removal specifications (LCO 3.4.9, LCO 3.4.10, LCO 3.9.8, and LCO 3.9.9) require subsystem(s) to be operable and one subsystem "to be in operation." These LCOs contain LCO Notes that exempt this requirement for a period of time. However, a subtle difference in the wording of these Notes as compared to the LCO words could cause confusion. These Notes say that the subsystem may be "removed from operation", which could imply that although the subsystem could be removed from operation it must be immediately placed back in operation to meet the LCO requirements. The same potential problem exists with the associated Bases, which use the term “be shut down.” The intent of the Notes is to allow the subsystem to “not be in operation” for a period of time. The subtle difference between the wording of the LCO requirement and the associated Notes adds potential confusion. This potential confusion is removed by making the LCO Notes consistent with the LCO being excepted. This is an administrative change, since it provides consistency throughout the Technical Specifications, and the original intent of the Notes is being maintained.

d. Technical Specifications Affected

	TS Pages	Bases Pages
LCO 3.4.9	3.4-21	B 3.4-45
LCO 3.4.10	3.4-24	B 3.4-50
LCO 3.9.8	3.9-10	B 3.9-26
LCO 3.9.9	3.9-13	B 3.9-31

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown

LCO 3.4.9 Two RHR shutdown cooling subsystems shall be OPERABLE, and, with no recirculation pump in operation, at least one RHR shutdown cooling subsystem shall be in operation.

---NOTES---

1. Both RHR shutdown cooling subsystems and recirculation pumps may ~~be removed from~~ not be in operation for up to 2 hours per 8 hour period.

2. One RHR shutdown cooling subsystem may be inoperable for up to 2 hours for performance of Surveillances.

All changes per Att. 11; TSTF 153

APPLICABILITY: MODE 3 with reactor steam dome pressure less than the RHR cut in permissive pressure.

ACTIONS

---NOTES---

1. LCO 3.0.4 is not applicable.

2. Separate Condition entry is allowed for each RHR shutdown cooling subsystem.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two RHR shutdown cooling subsystems inoperable.	A.1 Initiate action to restore RHR shutdown cooling subsystem(s) to OPERABLE status. AND	Immediately (continued)

Attachment 11
PY-CEI/NRR-2523L
Page 3 of 9

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Residual Heat Removal (RHR) Shutdown Cooling System—Cold Shutdown

LCO 3.4.10 Two RHR shutdown cooling subsystems shall be OPERABLE, and, with no recirculation pump in operation, at least one RHR shutdown cooling subsystem shall be in operation.

-----------------------------NOTES-----------------------------

1. Both RHR shutdown cooling subsystems and recirculation pumps may ~~be removed from~~ operation for up to 2 hours per 8 hour period.

not be in

2. One RHR shutdown cooling subsystem may be inoperable for up to 2 hours for the performance of Surveillances.
3. Both RHR shutdown cooling subsystems and recirculation pumps may ~~be removed from~~ operation during inservice leak and hydrostatic testing.

All changes per Att. 11;
& consistent with TSTF 153

APPLICABILITY: MODE 4, when heat losses to the ambient are not sufficient to maintain average reactor coolant temperature ≤ 200°F.

ACTIONS

-----------------------------NOTE-----------------------------

Separate Condition entry is allowed for each RHR shutdown cooling subsystem.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two RHR shutdown cooling subsystems inoperable.	A.1 Verify an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem.	1 hour AND Once per 24 hours thereafter

(continued)

Attachment 11
PY-CEI/NRR-2523L
Page 4 of 9

3.9 REFUELING OPERATIONS

3.9.8 Residual Heat Removal (RHR)—High Water Level

LCO 3.9.8 One RHR shutdown cooling subsystem shall be OPERABLE and in operation.

-----NOTE-----
The required RHR shutdown cooling subsystem may not be in ~~be removed from~~ operation for up to 2 hours per 8 hour period.

All changes per Att. 11; TSTF 153

APPLICABILITY: MODE 5 with irradiated fuel in the reactor pressure vessel (RPV) and with the water level ≥ 22 ft 9 inches above the top of the RPV flange, and heat losses to the ambient are not sufficient to maintain average reactor coolant temperature ≤ 140°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required RHR shutdown cooling subsystem inoperable.	A.1 Verify an alternate method of decay heat removal is available.	1 hour AND Once per 24 hours thereafter
B. Required Action and associated Completion Time of Condition A not met.	B.1 Suspend loading irradiated fuel assemblies into the RPV. AND	Immediately (continued)

3.9 REFUELING OPERATIONS

3.9.9 Residual Heat Removal (RHR)—Low Water Level

LCO 3.9.9 Two RHR shutdown cooling subsystems shall be OPERABLE, and one RHR shutdown cooling subsystem shall be in operation.

-----NOTE-----
The required operating shutdown cooling subsystem may ~~be~~ not be in ~~removed from~~ operation for up to 2 hours per 8 hour period.

All changes per Att. 11; TSTF 153

APPLICABILITY: MODE 5 with irradiated fuel in the reactor pressure vessel (RPV) and with the water level < 22 ft 9 inches above the top of the RPV flange, and heat losses to the ambient are not sufficient to maintain average reactor coolant temperature ≤ 140°F.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each RHR shutdown cooling subsystem.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or two RHR shutdown cooling subsystems inoperable.	A.1	Verify an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem.	1 hour <u>AND</u> Once per 24 hours thereafter
B. Required Action and associated Completion Time of Condition A not met.	B.1 <u>AND</u>	Initiate action to restore primary containment to OPERABLE status.	Immediately (continued)

BASES

LCO (continued)

or local) in the shutdown cooling mode for removal of decay heat. In MODE 3, one RHR shutdown cooling subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. Operation (either continuous or intermittent) of one subsystem can maintain or reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required.

All changes per Att. 11; TSTF 153

Note 1 permits both RHR shutdown cooling subsystems and recirculation pumps to not be in operation ~~be shut down~~ for a period of 2 hours in an 8 hour period. Note 2 allows one RHR shutdown cooling subsystem to be inoperable for up to 2 hours for performance of surveillance tests. These tests may be on the affected RHR System or on some other plant system or component that necessitates placing the RHR System in an inoperable status during the performance. This is permitted because the core heat generation can be low enough and the heatup rate slow enough to allow some changes to the RHR subsystems or other operations requiring RHR flow interruption and loss of redundancy.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with reactor steam dome pressure greater than or equal to the RHR cut in permissive pressure, this LCO is not applicable. Operation of the RHR System in the shutdown cooling mode is not allowed above this pressure because the RCS pressure may exceed the design pressure of the shutdown cooling piping. Decay heat removal at reactor pressures greater than or equal to the RHR cut in permissive pressure is typically accomplished by condensing the steam in the main condenser. Additionally, in MODE 2 below this pressure, the OPERABILITY requirements for the Emergency Core Cooling Systems (ECCS) (LCO 3.5.1, "ECCS—Operating") do not allow placing the RHR shutdown cooling subsystem into operation.

In MODE 3 with reactor steam dome pressure below the RHR cut in permissive pressure (i.e., the actual pressure at which the interlock resets) the RHR System may be operated in the shutdown cooling mode to remove decay heat to reduce or maintain coolant temperature. Otherwise, a recirculation pump is required to be in operation.

(continued)

Attachment 11
PY-CEI/NRR-2523L
Page 7 of 9

LCO (continued)

aligned (remote or local) in the shutdown cooling mode for removal of decay heat. In MODE 4, one RHR shutdown cooling subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. Operation (either continuous or intermittent) of one subsystem can maintain and reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required.

not be in operation

Note 1 permits both RHR shutdown cooling subsystems and recirculation pumps to ~~be shut down~~ for a period of 2 hours in an 8 hour period. Note 2 allows one RHR shutdown cooling subsystem to be inoperable for up to 2 hours for performance of surveillance tests. These tests may be on the affected RHR System or on some other plant system or component that necessitates placing the RHR System in an inoperable status during the performance. This is permitted because the core heat generation can be low enough and the heatup rate slow enough to allow some changes to the RHR subsystems or other operations requiring RHR flow interruption and loss of redundancy. Note 3 permits both RHR shutdown cooling subsystems and reactor recirculation pumps to ~~be shutdown~~ during performance of inservice leak testing and during reactor pressure vessel hydrostatic testing. This is permitted because RCS pressures and temperatures are being closely monitored during these tests as required by LCO 3.4.11, "RCS Pressure and Temperature (P/T) Limits."

not be in operation

All changes per Att. 11; & consistent with TSTF 153

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with reactor steam dome pressure greater than or equal to the RHR cut in permissive pressure, this LCO is not applicable. Operation of the RHR System in the shutdown cooling mode is not allowed above this pressure because the RCS pressure may exceed the design pressure of the shutdown cooling piping. Decay heat removal at reactor pressures greater than or equal to the RHR cut in permissive pressure is typically accomplished by condensing the steam in the main condenser. Additionally, in MODE 2 below this pressure, the OPERABILITY requirements for the Emergency Core Cooling Systems (ECCS) (LCO 3.5.1, "ECCS — Operating") do not allow placing the RHR shutdown cooling subsystem into operation.

(continued)

LCO (continued)

An RHR shutdown cooling subsystem is OPERABLE when the RHR pump, two heat exchangers in series, valves, piping, instrumentation, and controls are OPERABLE.

Additionally, each RHR shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. Operation (either continuous or intermittent) of one subsystem can maintain and reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required. A Note is provided to allow the required RHR shutdown cooling subsystem to ~~be removed from~~ not be in operation for up to two hours in an eight hour period.

All changes per ~~Att.~~ 11; TSTF 153

APPLICABILITY

One RHR shutdown cooling subsystem must be OPERABLE in MODE 5, with the water level ≥ 22 ft 9 inches above the top of the RPV flange, when heat losses to the ambient are not sufficient to maintain average reactor coolant temperature ≤ 140°F, to provide decay heat removal. Ambient losses must be such that no increase in reactor vessel water temperature will occur. With RPV water temperature remaining below 140°F, adequate margin is being maintained to coolant boiling, evaporative losses are minimal, and refueling floor environmental conditions will not be adversely affected. If temperature is not maintained below this value with only ambient heat losses, decay heat removal capability is required. RHR System requirements in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS); Section 3.5, Emergency Core Cooling Systems (ECCS) and Reactor Core Isolation Cooling (RCIC) System; and Section 3.6, Containment Systems. RHR Shutdown Cooling System requirements in MODE 5, with irradiated fuel in the RPV and with the water level < 22 ft 9 inches above the RPV flange, are given in LCO 3.9.9, "Residual Heat Removal (RHR)—Low Water Level."

ACTIONS

A.1

With no RHR shutdown cooling subsystem OPERABLE, an alternate method of decay heat removal must be established within 1 hour. In this condition, the volume of water above

(continued)

Attachment 11
PY-CEI/NRR-2523L
Page 9 of 9

BASES

LCO (continued)

All changes per Att. 11; TSTF 153

Additionally, each RHR shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. Operation (either continuous or intermittent) of one subsystem can maintain and reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required. A Note is provided to allow the required RHR shutdown cooling subsystem to ~~be removed from~~ not be in operation for up to two hours in an eight hour period.

APPLICABILITY

Two RHR shutdown cooling subsystems are required to be OPERABLE in MODE 5, with irradiated fuel in the RPV and with the water level < 22 ft 9 inches above the top of the RPV flange, when heat losses to the ambient are not sufficient to maintain average reactor coolant temperature ≤ 140°F, to provide decay heat removal. Ambient losses must be such that no increase in reactor vessel water temperature will occur. With RPV water temperature remaining below 140°F, adequate margin is being maintained to coolant boiling, evaporative losses are minimal, and refueling floor environmental conditions will not be adversely affected. If temperature is not maintained below this value with only ambient heat losses, decay heat removal capability is required. RHR System requirements in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS); Section 3.5, Emergency Core Cooling Systems (ECCS) and Reactor Core Isolation Cooling (RCIC) System; and Section 3.6, Containment Systems. RHR Shutdown Cooling System requirements in MODE 5, with irradiated fuel in the RPV and with the water level ≥ 22 ft 9 inches above the RPV flange, are given in LCO 3.9.8, "Residual Heat Removal (RHR) – High Water Level."

ACTIONS

A Note has been provided to modify the ACTIONS related to RHR shutdown cooling subsystems. Section 1.3, Completion Times, specifies once a Condition has been entered, subsequent divisions, subsystems, components or variables expressed in the Condition, discovered inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable

(continued)

TSTF 166 Correct Inconsistency Between LCO 3.0.6 and the SFDP Regarding Performance of an Evaluation

a. Requested Change

Revise LCO 3.0.6 to explicitly require an evaluation per the Safety Function Determination Program (SFDP). Delete statement "additional . . . limitations may be required" from LCO 3.0.6.

b. Consistency with TSTF

The proposed change is consistent with the TSTF.

c. Justification

There is an inconsistency between LCO 3.0.6, the LCO 3.0.6 Bases, and the SFDP. This change resolves the inconsistency. As currently written, LCO 3.0.6 does not explicitly require an evaluation in accordance with the SFDP, rather it states that additional evaluations may be required. Both the SFDP and the LCO 3.0.6 Bases state that upon entry into LCO 3.0.6, an evaluation shall be made to determine if a loss of safety function exists. In addition, because LCO 3.0.6 states that the evaluation be done in accordance with the SFDP and the SFDP states that other appropriate actions may be taken, there is no need for the statement "additional . . . limitations may be required" in LCO 3.0.6. This is an administrative change, since it provides consistency throughout the Technical Specifications, and the original intent of LCO 3.0.6 and the SFDP are being maintained.

d. Technical Specifications Affected

	TS Pages	**Bases Pages**
LCO 3.0.6	3.0-2	None

Attachment 12
PY-CEI/NRR-2523L
Page 2 of 2

3.0 LCO APPLICABILITY

LCO 3.0.4 (continued)

specified conditions in the Applicability that are required to comply with ACTIONS, or that are part of a shutdown of the unit.

Exceptions to this Specification are stated in the individual Specifications. ~~These exceptions allow entry into MODES or other specified conditions in the Applicability when the associated ACTIONS to be entered allow unit operation in the MODE or other specified condition in the Applicability only for a limited period of time.~~

This change per Att. 7; TSTF 104

LCO 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, and 3.

LCO 3.0.5

Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LCO 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.

LCO 3.0.6

When a supported system LCO is not met solely due to a support system LCO not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system LCO ACTIONS are required to be entered. This is an exception to LCO 3.0.2 for the supported system. In this event, an ~~additional~~ evaluation ~~and limitations may be required~~ shall be performed in accordance with Specification 5.5.10, "Safety Function Determination Program (SFDP)." If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

This change per Att. 12; TSTF 166

When a support system's Required Action directs a supported system to be declared inoperable or directs entry into Conditions and Required Actions for a supported system, the applicable Conditions and Required Actions shall be entered in accordance with LCO 3.0.2.

(continued)

TSTF 258 Changes to Section 5.0, Administrative Controls

a. Requested Change

There are a number of separate changes to Section 5 proposed within this TSTF. Only a portion of the changes are requested for PNPP at this time. The portions being requested are the administrative/editorial/consistency changes that are not already addressed in the PNPP Technical Specifications. Most notably, the TSTF included changes to Administrative Control 5.7 "High Radiation Area", which are not being requested, since the changes could be considered technical in nature.

The changes requested are:
1. Delete Section 5.2.2.b, which is redundant to 10CFR50.54 requirements.
2. Add a new Section 5.3.2, to clarify compliance with 10CFR55.4 requirements.
3. Revise Section 5.5.4 to be consistent with the intent of 10CFR20:
   a. revise 5.5.4.g and j to clarify where dose rate limits apply at, with respect to the site boundary
   b. revise 5.5.4.g to change the phrase "total body" to "whole body" for noble gas dose rate
   c. add a new paragraph below 5.5.4.j, to note that SR 3.0.2 and 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.
4. Remove unnecessary reporting requirement on lifts of safety/relief valves from Section 5.6.4.

b. Consistency with TSTF

The proposed change is consistent with the TSTF, with the following exceptions:

- When paragraphs are deleted, the subsequent paragraphs are not re-numbered. Re-numbering leads to unnecessary procedure changes.
- Changes to Section 5.2.2.e on unit staff working hours are not necessary, since Section 5.2.2.e working hour rules were revised for PNPP by Amendment 98. Therefore, these changes (including inserts A and G), are not incorporated.
- Changes to Section 5.2.2.g on STA qualifications are not necessary, since the intent of the TSTF changes are met by an existing sentence in PNPP Section 5.2.2.g which states "The STA position may be filled by an on-shift SS or SRO provided the individual meets the Commission Policy Statement on Engineering Expertise On Shift".
- The 10CFR20 dose limits in TSTF-258 Inserts C and D are already incorporated in PNPP Section 5.5.4 "Radioactive Effluent Controls Program". Therefore, these two inserts are not incorporated.
- Insert F for Section 5.7 "High Radiation Area" is not incorporated, because the changes could be considered to be technical in nature, and would not be within the scope of this administrative license amendment request.

c. Justifications for each of the subitems listed above:

1. The requirements of 10CFR50.54(m)(2)(iii) and 50.54(k) adequately provide for shift manning. 50.54(m)(2)(iii) requires "When a nuclear power unit is in an

operational mode other than cold shutdown or refueling, as defined by the unit's technical specifications, each licensee shall have a person holding a senior operator license for the nuclear power unit in the control room at all times. In addition to this senior operator, for each fueled nuclear power unit, a licensed operator or senior operator shall be present at the controls at all times." Further, 50.54(k) requires "An operator or senior operator licensed pursuant to Part 55 of this chapter shall be present at the controls at all times during the operation of the facility." The requirements in Section 5.2.2.b will be met through compliance with these regulations, and the requirements are not required to be reiterated in the Technical Specifications.

2. Section 5.3.2 is added to ensure that there is no misunderstanding when complying with 10CFR55.4 requirements. The Definitions in 10CFR55.4 state "*Actively performing the functions of an operator or senior operator* means that an individual has a position on the shift crew that requires the individual to be licensed **as defined in the facility's technical specifications**..." (bolding is added; italics are from 10CFR55.4). To give the pointer in the Definition a target within the Technical Specifications, a sentence is added stating "For the purpose of 10CFR55.4, a licensed Senior Reactor Operator (SRO) and a licensed operator (RO) are those individuals who, in addition to meeting the requirements of TS 5.3.1, perform the functions described in 10CFR50.54(m)."

3.a. The changes to 5.5.4.g and j are simply editorial changes clarifying whether dose rates/doses from the site are measured "at or beyond" the site boundary (5.5.4.g), or just "beyond" the site boundary (5.5.4.j). The changes to 5.5.4.g are consistent with the definition of Unrestricted Area (which is the term used in 10CFR50 Appendix I). The changes to 5.5.4.j are consistent with the definition of general environment in 40CFR190. These wording changes are clarifications, and do not change any existing requirements or methods for measurement of dose rate/dose.

3.b. The PNPP Technical Specifications currently use the term "total body" in item 5.5.4.g, in reference to the noble gas dose rate. The limit is based on the dosimetry of ICRP 2 and the correct term is "whole body" as shown in NUREG-1302 "Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Boiling Water Reactors", Specification 3.11.2.1, page 46 (Note: the TSTF references NUREG-1301, which is N/A to PNPP since -1301 is for pressurized water reactors). This wording change is a clarification, and does not change any existing requirements or methods for measurement of dose rate.

3.c. As noted above for TSTFs 52 and 118, the NRC and the TSTF extensively debated whether the Section 3.0 provisions (such as SR 3.0.2) generically apply to the Section 5 Administrative Controls. The NRC position was that Section 3.0 does apply to Chapter 5. In a meeting on 12/15/98 and in a letter dated June 28, 1999 from W. Beckner (NRC) to J. Davis (NEI), the staff stated "It is the staff position that SR 3.0.2 does apply to the frequencies that are explicitly stated in Section 5.0... or encompassed within a special program (i.e., "Radioactive Effluent Controls Program)...". However, the final decision was that Section 3.0 does not generically apply to Chapter 5. Based on this

final agreement, Administrative Control 5.5.4 is revised to add the following sentence, "The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency." This change is an administrative clarification only, since it provides consistency with the requirements for other programs in Section 5.0, and it maintains the original intent of the NRC by applying SR 3.0.2 and SR 3.0.3 to this program.

4. The original reporting of safety/relief valve challenges is based on guidance in NUREG-0694 "TMI-Related Requirements for New Operating Licenses". The guidance of NUREG-0694 states: "All challenges to the ...safety valves should be documented in the annual report." As part of the PNPP conversion to the iSTS, this requirement was changed from being part of the Annual Report to the Monthly Operating Report (Section 5.6.4). Subsequently, NRC Generic Letter 97-02 "Revised Contents of the Monthly Operating Report" requests the submittal of less information in the monthly operating report. The Generic Letter identifies what needs to be reported to support the NRC Performance Indicator Program. The Generic Letter does not identify the need to report challenges to the safety/relief valves. Mr. Marcel Harper, NRC (AEOD) (before AEOD was disbanded) was contacted and he indicated that this information was not required for the Performance Indicator Program and therefore would not need to be reported. Also, TSTF-258 (this TSTF), which proposed deletion of the report, was approved by NRC letter dated 6/29/99. Based on this information, it is acceptable to delete the administrative requirement to provide a report of all challenges to safety/relief valves.

d. Technical Specifications Affected

	TS Pages	**Bases Pages**
Administrative Control 5.2.2	5.0-3	N/A
Administrative Control 5.3.2	5.0-4	
Administrative Control 5.5.4	5.0-9	
Administrative Control 5.6.4	5.0-17	

Attachment 13
PY-CEI/NRR-2523L
Page 4 of 9

5.2.2 Unit Staff

The unit staff organization shall include the following:

a. A non-licensed operator shall be on site when fuel is in the reactor vessel, and an additional non-licensed operator shall be on site while the unit is in MODE 1, 2, or 3.

This change per Att. 13; TSTF 258

b. Deleted ~~At least one licensed Reactor Operator (RO) shall be present in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, or 3, at least one licensed Senior Reactor Operator (SRO) shall be present in the control room.~~

c. Shift crew composition may be one less than the minimum requirements of 10 CFR 50.54(m)(2)(i) and Specifications 5.2.2.a and 5.2.2.g for a period of time not to exceed two hours in order to accommodate unexpected absence of on-duty shift crew members, provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

These changes per Att. 6; & consistent with TSTF 65

d. A ~~health physics~~ radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

e. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety related functions (e.g., licensed SROs, licensed ROs, ~~health physicists~~ radiation protection technicians, auxiliary operators, and key maintenance personnel). The procedures shall include guidelines on working hours that ensure that adequate shift coverage is maintained without routine heavy use of overtime.

Any deviation from the working hour guidelines shall be authorized in advance by the Plant Manager or his designees, in accordance with approved administrative procedures, or by higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation.

Controls shall be included in the procedures such that the individual overtime shall be reviewed monthly by the Plant Manager or his designees to ensure that excessive hours have not been assigned. Routine deviation from the working hour guidelines is not authorized.

(continued)

5.2.2 Unit Staff (continued)

f. The operations manager or at least one operations middle manager shall hold an SRO license.

g. The shift technical advisor (STA) shall provide advisory technical support to the shift supervisor (SS) in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit.

In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on shift. The STA position may be filled by an on-shift SS or SRO provided the individual meets the Commission Policy Statement on Engineering Expertise on shift.

No changes to this page;
FYI for Att. 13; TSTF 258

5.0 ADMINISTRATIVE CONTROLS

Attachment 13
PY-CEI/NRR-2523L
Page 6 of 9

5.3 Unit Staff Qualifications

5.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions as modified by Specification 5.2.2.f, except for the radiation protection manager, who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and the licensed Reactor Operators and Senior Reactor Operators, who shall comply with the requirements of 10 CFR 55.

5.3.2 For the purpose of 10 CFR 55.4, a licensed Senior Reactor Operator (SRO) and a licensed reactor operator (RO) are those individuals who, in addition to meeting the requirements of TS 5.3.1, perform the functions described in 10 CFR 50.54(m).

All changes per Att. 13;
TSTF 258

5.5.4 Radioactive Effluent Controls Program

This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas, conforming to ten times the concentration values in 10 CFR 20, Appendix B, Table 2, Column 2;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from the unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current quarter and current year in accordance with the methodology and parameters in the ODCM at least every 31 days;

(continued)

No changes to this page;
provided for information

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary as follows:

1. for noble gases: ≤ 500 mrem/yr to the ~~total~~ whole body and ≤ 3000 mrem/yr to the skin, and

2. for iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives > 8 days: ≤ 1500 mrem/yr to any organ;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from the unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from the unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the USAR, Section 3.9.1.1, cyclic and transient occurrences to ensure that the reactor vessel is maintained within the design limits.

(continued)

All changes are per Att. 13; TSTF 258

Attachment 13
PY-CEI/NRR-2523L
Page 9 of 9

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the unit during the previous year shall be submitted by May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and process control program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, ~~including documentation of all challenges to the main steam safety/relief valves,~~ shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 Core Operating Limits Report (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1. LCO 3.2.1, Average Planar Linear Heat Generation Rate (APLHGR),

2. LCO 3.2.2, Minimum Critical Power Ratio (MCPR),

3. LCO 3.2.3, Linear Heat Generation Rate (LHGR), and

(continued)

All changes per Att. 13; TSTF 258

TSTF 278 Battery Cell Parameters (LCO 3.8.6) includes more than Table 3.8.6-1 limits

a. Requested Change

Specification 3.8.6 “Battery Cell Parameters” contains requirements in two places (in Table 3.8.6-1 and in its surveillance requirements). Therefore, LCO 3.8.6 is revised to require that battery cell parameters be "within limits" rather than “within the limits of Table 3.8.6-1.” Additionally, editorial changes are made to the Actions Table to make the references to the Table 3.8.6-1 Category A, B, and C limits consistent.

b. Consistency with TSTF

The proposed change is consistent with the TSTF with the following clarifications:

- The TSTF adds a reference in Condition A to Table 3.8.6-1; this reference is already included in PNPP Condition A.

c. Justification

LCO 3.8.6 requires the cell parameters for the batteries to be within the limits specified in Table 3.8.6-1. This requirement is not inclusive of all the limits specified in the applicable SRs. In addition to the limits specified in Table 3.8.6-1, a limit regarding the average electrolyte temperature is contained within an SR (SR 3.8.6.3). Therefore, the LCO is in conflict with the SR requirements. LCO 3.8.6 is revised to require the battery cell parameters to be within limits. This change resolves the conflict between the LCO and SRs.

Additionally, the references in the Actions Table to the Table 3.8.6-1 Category A, B, and C limits were made consistent. These are purely editorial changes.

This is an administrative change, since these changes do not impact the requirements of the Technical Specifications.

d. Technical Specifications Affected

	TS Pages	Bases Pages
LCO 3.8.6	3.8-32 3.8-33	None

3.8 ELECTRICAL POWER SYSTEMS

3.8.6 Battery Cell Parameters

LCO 3.8.6 Battery cell parameters for the Division 1, 2, and 3 batteries shall be within ~~the~~ limits. ~~of Table 3.8.6-1.~~

APPLICABILITY: When associated DC electrical power subsystems are required to be OPERABLE.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each battery.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more batteries with one or more battery cell parameters not within Table 3.8.6-1 Category A or B limits.	A.1 Verify pilot cell's electrolyte level and float voltage meet Table 3.8.6-1 Category C limits.	1 hour
	AND	
	A.2 Verify battery cell parameters meet Table 3.8.6-1 Category C limits.	24 hours AND Once per 7 days thereafter
	AND	
	A.3 Restore battery cell parameters to Category A and B limits ~~of~~ Table 3.8.6-1.	31 days

(continued)

All changes per Att. 14; TSTF 278

Attachment 14
PY-CEI/NRR-2523L
Page 3 of 4

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR One or more batteries with average electrolyte temperature of the representative cells < 72°F. OR One or more batteries with one or more battery cell parameters not within Table 3.8.6-1 Category C limits.	B.1 Declare associated battery inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.6.1 Verify battery cell parameters meet Table 3.8.6-1 Category A limits.	7 days

(continued)

All changes per Att. 14; TSTF 278

Attachment 14
PY-CEI/NRR-2523L
Page 4 of 4

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.6.2	Verify battery cell parameters meet Table 3.8.6-1 Category B limits.	92 days AND Once within 72 hours after battery overcharge > 145 V
SR 3.8.6.3	Verify average electrolyte temperature of representative cells is ≥ 72°F.	92 days

FYI

No changes to this page; information only for Att. 14; TSTF 278

TSTF 279 Remove the words "including applicable supports" from the description of the Inservice Testing Program

a. Requested Change

Delete the wording "including applicable supports" from the description of the "Inservice Testing Program" contained in Section 5.5.6.

b. Consistency with TSTF

The proposed change is consistent with the TSTF.

c. Justification

The Inservice Testing (IST) Program provides controls for testing Code Class 1, 2 and 3 components. The Inservice Examination (ISE) Program addresses items such as piping welds and pipe supports. The discussion of the IST Program in Section 5.5.6 of the iSTS was revised by the NRC to include the words "including applicable supports" in February 1992 due to issues related to the relocation of the snubber LCO from the iSTS NUREGs. However, this was inappropriate since supports are addressed under the Inservice Examination (ISE) Program, not the IST Program. Thus, the reference to the applicable supports in the IST Program description in Section 5.5.6 is deleted.

In the last six years, many plants have implemented iSTS with no issues related to testing of snubbers or inspection of supports. The American Society of Mechanical Engineers (ASME) has developed OM-5 and other guidance to provide appropriate requirements for supports and snubbers. Since supports are addressed under the ISE Program, not the IST Program, these changes do not impact the requirements of the Technical Specifications, and this is considered to be an administrative change.

d. Technical Specifications Affected

	TS Pages	Bases Pages
Administrative Control 5.5.6	5.0-10	None

Attachment 15
PY-CEI/NRR-2523L
Page 2 of 2

5.5.6 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components~~, including applicable supports~~. The program shall include the following:

All changes per Att. 15 TSTF 279

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

FYI

b. The provisions of SR 3.0.2 are applicable to the above required frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

5.5.7 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in Regulatory Guide 1.52, Revision 2.

(continued)

SIGNIFICANT HAZARD CONSIDERATION

Changes are proposed to the current Perry Nuclear Power Plant (PNPP) Technical Specifications to incorporate many of the generic improvements agreed upon between the Industry and the NRC since the PNPP conversion to the improved Technical Specifications.

These changes involve reformatting and rewording of Technical Specifications to be consistent with regulations or other existing Technical Specifications, or the changes do not involve a change in intent. As a result, they are considered to be administrative in nature. The PNPP Technical Specifications are being revised to adopt NRC approved TSTF-5, TSTF-32, TSTF-38, TSTF-52, TSTF-65, TSTF-104, TSTF-106, TSTF-118, TSTF-152, TSTF-153, TSTF-166, TSTF-258, TSTF-278, and TSTF-279, which provide generic changes to the improved Standard Technical Specifications (iSTS) as outlined in NUREG-1434, "Standard Technical Specifications, BWR/6 Plants."

The standards used to arrive at a determination that a request for amendment does not involve a significant hazard are included in Commission regulation 10CFR50.92, which states that operation of the facility in accordance with the proposed changes would not:

1) involve a significant increase in the probability or consequences of an accident previously evaluated; or
2) create the possibility of a new or different kind of accident from any accident previously evaluated; or
3) involve a significant reduction in a margin of safety.

The proposed amendment has been reviewed with respect to these three factors, and it has been determined that the proposed change does not involve a significant hazard because:

This proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes involve reformatting and rewording of the existing Technical Specifications to be consistent with regulations or other existing Technical Specifications, or the changes do not involve a change in intent. The proposed changes also involve Technical Specification requirements that are administrative rather than technical in nature. As such, this change does not affect initiators of previously evaluated events, or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

This proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed changes will not impose new or eliminate old requirements on design or operation of the plant. The administrative changes also do not introduce new initiators of events. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

This proposed amendment does not involve a significant reduction in a margin of safety.

The proposed change has no impact on any safety analysis assumptions or design basis margins. This change is administrative in nature. The proposed changes will not impose new or eliminate old requirements on design or operation of the plant. Therefore, the change does not involve a significant reduction in a margin of safety.

Based on the above considerations, it is concluded that a significant hazard would not be introduced as a result of this proposed change. Also, since NRC approval of this change must be obtained prior to implementation, no unreviewed safety question can exist.