Florida Power & Light Company, 6501 South Ocean Drive, Jensen Beach, FL 34957

FPL

April 17, 2001

L-2001-078
10 CFR 50.90

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

RE: St. Lucie Units 1 and 2
Docket Nos. 50-335 and 50-389
Proposed License Amendments
Minor Changes/Corrections

Pursuant to 10 CFR 50.90, Florida Power and Light Company (FPL) requests to amend Facility Operating Licenses DPR-67 for St. Lucie Unit 1 and NPF-16 for St. Lucie Unit 2 by incorporating the attached Technical Specifications (TS) revisions. The proposed amendments are required to correct various minor errors or incorporate 10 CFR 50.59 Rule conforming amendments within the existing Technical Specifications.

Attachment 1 is an evaluation of the proposed changes. Attachment 2 is the "Determination of No Significant Hazards Consideration." Attachments 3 and 4 contain copies of the affected Technical Specifications pages marked up to show the proposed changes.

The St. Lucie Facility Review Group and the FPL Company Nuclear Review Board have reviewed the proposed amendments. In accordance with 10 CFR 50.91 (b) (1), copies of the proposed amendments are being forwarded to the State Designee for the State of Florida.

There is no particular requested need date for this licensing action request. Please Issue the amendment to be effective on the date of issuance and to be implemented within 60 days of receipt by FPL. Please contact us if there are any questions about this submittal.

Very truly yours,

R. S. Kundalkar
Vice President
St. Lucie Plant

RSK/EJW/KWF

Attachments

cc: Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, St. Lucie Plant
Mr. W. A. Passetti, Florida Department of Health and Rehabilitative Services

A001

STATE OF FLORIDA)
) ss.
COUNTY OF ST. LUCIE)

R. S. Kundalkar being first duly sworn, deposes and says:

That he is Vice President, St. Lucie Plant, for the Nuclear Division of Florida Power and Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this document are true and correct to the best of his knowledge, information and belief, and that he is authorized to execute the document on behalf of said Licensee.

Rajiv S. Kundalkar
R. S. Kundalkar

STATE OF FLORIDA

COUNTY OF St. Lucie

Sworn to and subscribed before me

this 17 day of April, 2001

by R. S. Kundalkar, who is personally known to me.

Leslie J. Whitwell
Signature of Notary Public-State of Florida

Leslie J. Whitwell
MY COMMISSION # CC646183 EXPIRES
May 12, 2001
BONDED THRU TROY FAIN INSURANCE, INC.

Name of Notary Public (Print, Type, or Stamp)

EVALUATION OF PROPOSED TS CHANGES

EVALUATION OF PROPOSED TS CHANGES

Introduction

Florida Power and Light Company (FPL) requests to amend Facility Operating Licenses DPR-67 for St. Lucie Unit 1 and NPF-16 for St. Lucie Unit 2 by incorporating the attached Technical Specifications (TS) revisions. The proposed amendments are required to correct various minor errors or incorporate 10 CFR 50.59 Rule conforming amendments within the existing Technical Specifications.

Background/Discussion/Justification for Change

The background and justification for the changes are discussed below. TS page mark ups for the proposed changes are in Attachments 3 and 4.

A. Unit 1 Technical Specification Corrections

1. License Condition 2.C(3) has two typographical errors in that on page 3 FPL letter L-83-227 should be dated April "12," 1983, not April "22," 1983, and that on page 4 an NRC letter is incorrectly identified as October 4, "1998" instead of "1988."

   Therefore, License Condition 2.C(3) will be revised such that "L-83-227 dated April 22, 1983," will be changed to "L-83-227 dated April 12, 1983," and the NRC letter identified as "October 4, 1998," will be changed to "October 4, 1988."

2. TS page 3/4 3-15, Table 3.3-4, "Engineered Safety Feature Actuation System Instrumentation Trip Values," has an error in the Amendment No. listing on the bottom of the page. TS amendment 72, not 74, changed this TS table.

   Therefore TS Table 3.3-4 will be revised to change the third amendment number listed on the bottom of the page that reads "~~74~~," to "72."

3. The Limiting Condition for Operation (LCO) for TS 3.3.3.1, TS Page 3/4 3-21, Radiation Monitoring, states that "The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE* with their alarm setpoints within the specified limits. *The emergency power source may be inoperable in Modes 5 and 6." Therefore, in order for a radiation monitor, capable of being powered from an emergency power source, to be considered OPERABLE in Modes 1 though 4 the LCO implies that the radiation monitor must have an operable EDG. With the exception of the containment area radiation monitors, the radiation detectors listed in TS Table 3.3-6 are essentially single train quality related systems, in that redundant capability is not provided by

design. The containment area radiation monitors are part of a safety related, redundant system used for initiating the containment isolation signal (CIS).

However, the asterisked statement within the LCO for TS 3.3.3.1 is no longer applicable and should have been removed by a previous license amendment. The asterisked statement in the LCO was added to TS 3.3.3.1 as part of TS amendment 40 that added TS 3.0.5 to the St. Lucie Unit 1 TS. TS amendment 40 also changed the definition of OPERABILITY such that normal and emergency electrical power was required to consider a component OPERABLE.

In 1985, St. Lucie Unit 1 TS amendment 69 was implemented. Among other changes, this TS revised the definition section of the St. Lucie Unit 1 TS to be consistent with the TS being developed for Unit 2 as part of its initial licensing efforts. TS amendment 69 dropped the distinction between normal and emergency power requirements by redefining OPERABILITY such that only electrical power was required to consider a component OPERABLE.

In 1989, TS amendment 103 was approved that pertained to EDG operability requirements. TS 3.0.5 was deleted and the EDG operability and support system requirements were clarified and moved within TS 3.8.1.1. As discussed in the TS BASES for section 3/4.8, Electrical Power Systems, when one EDG is inoperable, there is an additional action requirement to verify that all required systems, subsystems, trains, components, and devices are operable that depend on the remaining operable EDG to be operable. The NUREG-1432 basis for this opposite train verification is to provide assurance that a loss of offsite power, during the period that a EDG is inoperable, does not result in a complete loss of critical systems. Safety related radiation monitors that provide critical functions have redundancy and are subject to the LCOs for TS 3.8.1.1. However, there is no need to declare non-critical single train quality related radiation monitors supported by the inoperable EDG inoperable.

The St. Lucie Unit 1 submittal for TS amendment 103 should have deleted the asterisk and associated statement "**The emergency power source may be inoperable in Modes 5 and 6.*" Redundant safety related radiation monitor operability is assured during the opposite train verification should an EDG become inoperable. Although many of the quality related single train radiation monitors are capable of being powered from an emergency power source, they do not provide critical functions requiring an operable emergency power backup. The LCO for St. Lucie Unit 2 TS 3.3.3.1 does not have the asterisked statement but is otherwise identical to the Unit 1 TS, so this change also brings consistency between the units.

Therefore, FPL proposes to delete the asterisked statement in the LCO for TS 3.3.3.1.

4. TS page 3/4 3-42, Table 3.3-11, "Accident Monitoring Instrumentation," incorrectly lists three channels of pressurizer water level instrumentation under the Total No. of Channels column. NUREG-1432, "Standard Technical Specifications for CE Plants," only requires two pressurizer level channels. Pressurizer level channels LI-1110X and LI-1110Y are environmentally qualified and credited to satisfy Regulatory Guide 1.97 instrumentation requirements. The third channel, LI-1103, is not environmentally qualified, is not powered by Class 1E power, and is not credited to satisfy Regulatory Guide 1.97 requirements. This change is consistent with the BASES for TS 3/4.3.3.8 that requires that instrumentation is available during and after an accident to assess plant conditions.

   Therefore, TS Table 3.3-11 will be revised on page 3/4 3-42 to read "2" instead of "3" Total No. of Channels of Pressurizer Water Level Instrumentation.

5. The TS headings for TS 3.5.2 and 3.5.3 need to be corrected. NUREG-1432 labels TS 3.5.2 as "ECCS – Operating" and TS 3.5.3 as "ECCS – Shutdown." The St. Lucie TS label these sections as "ECCS Subsytems – $T_{ave} \geq 325^{o}F$" and "ECCS Subsytems – $T_{ave} < 325^{o}F$." The current TS are incorrect in that Mode 3 requirements for ECCS subsystems exist in TS 3.5.3, but Mode 3 is defined as $T_{ave} \geq 325^{o}F$. The correct breakpoint for TS 3.5.2 or TS 3.5.3 is not temperature related, but depends on whether or not pressurizer pressure is greater than or less than 1750 psia. The existing TS pressurizer pressure breakpoints are clearly established by the Mode 3 asterisk and remain unchanged.

   Therefore, the heading for TS 3.5.2 (page 3/4 5-3) shall be changed to "ECCS Subsystems – Operating" and the heading for TS 3.5.3 (page 3/4 5-7) shall be changed to "ECCS Subsystems – Shutdown."

6. TS amendment 115 deleted 6.2.2.e from the Unit Staff Organization from the Technical Specification administrative requirements section. This requirement for the Fire Brigade was relocated out of the TS. However, the TS 6.2.2 sub bullets were incorrectly updated when the amendment was implemented.

   Therefore TS 6.2.2, page 6-2 will add "e. Deleted."

7. TS page 6-4 has an error in the Amendment No. listing on the bottom of the page. TS amendment 51 not 57, changed this TS page.

   Therefore, TS page 6-4 will be revised to change the second amendment number listed on the bottom of the page that reads "~~57~~," to "~~51~~."TS.

8. TS page 6-14 has an error in the Amendment No. listing on the bottom of the page. TS amendment 126, not 125, changed this TS page.

   Therefore, TS page 6-14 will be revised to change the second amendment number listed on the bottom of the page that reads "~~125~~," to "~~126~~."TS.

9. TS page 6-15c is being revised by FPL submittal L-2001-13, dated January 17, 2001. However, the NRC reviewer proposed different wording based on a future revision to TSTF-364, "Revisions to TS Bases Control Program to Incorporate Changes to 10 CFR 50.59."

   Therefore per TSTF-364, as subsequently revised by WOG-ED-24, "involve" will be changed to "require" for the new 6.8.4.j proposed under FPL submittal L-2001-13.

10. TS page 6-23 has a typographical error in that "ODCM" is spelled as "OCDM."

   Therefore, the "Changes to the OCDM" on page 6-23 of TS 6.14 shall be changed to read "Changes to the ODCM."

B. Unit 2 Technical Specification Corrections

1. TS amendment 109 changed TS Table 3.3-10, "Accident Monitoring Instrumentation," to change the reactor coolant outlet temperature - T_{Hot} instrument from "Narrow" to "Wide" range instrumentation. The reason the change was acceptable was that the wide range instrumentation is credited for meeting Regulatory Guide 1.97 instrumentation and is environmentally qualified, whereas the narrow range instrumentation is not. However, TS Table 4.3-7, "Accident Monitoring Instrumentation Surveillance Requirements," was not changed to reflect the correct instrument for reactor coolant outlet temperature - T_{Hot}.

   Therefore, on page 3/4 3-43, TS Table 4.3-7 for the reactor coolant outlet temperature - T_{Hot} shall be changed from "(Narrow Range)" to "(Wide Range)."

2. The TS headings for TS 3.5.2 and 3.5.3 need to be corrected. NUREG-1432 labels TS 3.5.2 as "ECCS – Operating" and TS 3.5.3 as "ECCS – Shutdown." The St. Lucie TS label these sections as "ECCS Subsytems – T_{ave} Greater Than or Equal to 325°F" and "ECCS Subsytems – T_{ave} Less Than 325°F." The current TS are incorrect in that Mode 3 requirements for ECCS subsystems exist in TS 3.5.3, but Mode 3 is defined as $T_{ave} \geq$ 325°F. The correct breakpoint for TS 3.5.2 or TS 3.5.3 is not temperature related, but depends on whether or not pressurizer pressure is greater than or less than 1750 psia. The existing TS pressurizer pressure breakpoints are clearly established by the Mode 3 asterisk and remain unchanged.

Therefore, the heading for TS 3.5.2 (page 3/4 5-3) shall be changed to "ECCS Subsystems – Operating" and the heading for TS 3.5.3 (page 3/4 5-7) shall be changed to "ECCS Subsystems – Shutdown."

3. TS page 6-15c is being revised by FPL submittal L-2001-13, dated January 17, 2001. However, the NRC reviewer proposed different wording based on a future revision to TSTF-364, "Revisions to TS Bases Control Program to Incorporate Changes to 10 CFR 50.59."

   Therefore per TSTF-364, as subsequently revised by WOG-ED-24, "involve" will be changed to "require" for the new 6.8.4.j proposed under FPL submittal L-2001-13.

C. Units 1 and 2 TS Conforming Changes to Implement Changed 10 CFR 50.59 Rule.

Recently a rule change became effective for 10 CFR 50.59. As a result, several terms used in the St. Lucie Unit 1 and 2 TS are obsolete (e.g., safety evaluation, unreviewed safety question, etc.) and need to be revised to reflect the new rule. These changes are required for the administrative TS governing the Facility Review Group (FRG), the Company Nuclear Review Board (CNRB), and procedure changes for both St. Lucie Units 1 and 2. The proposed wording is similar to that proposed in TSTF-364, Revision 0, "Revisions to TS Bases Control Program to Incorporate Changes to 10 CFR 50.59."

1. Unit 1 page 6-7 substitute "50.59" for "safety" in 6.5.1.6.a.

2. Unit 1 page 6-8 substitute "requires NRC approval pursuant to 10 CFR 50.59" for "constitutes an unreviewed safety question" in 6.5.1.7.b.

3. Unit 1 page 6-10 delete "safety" and substitute "require NRC approval pursuant to 10 CFR 50.59" for "constitute an unreviewed safety question" in 6.5.2.7.a, substitute "require NRC approval pursuant to 10 CFR 50.59" for "involve an unreviewed safety question as defined in Section 50.59, 10 CFR" in 6.5.2.7.b, and substitute "require NRC approval pursuant to 10 CFR 50.59" for "involve an unreviewed safety question as defined in Section 50.59, 10 CFR" in 6.5.2.7.c.

4. Unit 1 page 6-14 delete "safety" in 6.8.3.a.

5. Unit 2 page 6-8 substitute "50.59" for "safety" in 6.5.1.6.a.

6. Unit 2 page 6-9 substitute "requires NRC approval pursuant to 10 CFR 50.59" for "constitutes an unreviewed safety question" in 6.5.1.7.b.

7. Unit 2 page 6-11 delete "safety" and substitute "require NRC approval pursuant to 10 CFR 50.59" for "constitute an unreviewed safety question" in 6.5.2.7.a, substitute "require NRC approval pursuant to 10 CFR 50.59" for "involve an unreviewed safety question as defined in Section 50.59, 10 CFR" in 6.5.2.7.b, and substitute "require NRC approval pursuant to 10 CFR 50.59" for "involve an unreviewed safety question as defined in Section 50.59, 10 CFR" in 6.5.2.7.c.

8. Unit 2 page 6-14 delete "safety" from 6.8.3.a.

Conclusion

As discussed above, these proposed changes to the TS are needed to correct errors that are either administrative in nature, should have been included in previously approved licensing actions, or are required to support the new 10 CFR 50.59 rule change.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

Description of amendment request: The proposed license amendments (PLAs) to Facility Operating Licenses DPR-67 for St. Lucie Unit 1 and NPF-16 for St. Lucie Unit 2 are necessary to either correct existing minor errors within the Technical Specifications or implement conforming amendments to support the revised 10 CFR 50.59 rule.

Pursuant to 10 CFR 50.92, a determination may be made that a proposed license amendment involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (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. Each standard is discussed as follows.

(1) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

These proposed license amendments require no plant hardware or operational modifications. The proposed changes either correct various administrative errors (e.g., typographical errors, amendment tracking number errors), incorporate changes that have been justified by previously approved license amendments and should have been made as part of those submittals, correct logic errors, or are necessary to implement the 10 CFR 50.59 rule change. Therefore, operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

No modifications to either plant hardware or operational procedures are required to support these proposed license amendments; hence, no new failure modes are created. The proposed changes either correct various administrative errors (e.g., typographical errors, amendment tracking number errors), incorporate changes that have been justified by previously approved license amendments and should have been made as part of those submittals, correct logic errors, or are necessary to implement the 10 CFR 50.59 rule change. Therefore, operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

The majority of TS corrections proposed by these license amendments are administrative in nature in that they either correct typographical errors (e.g., ODCM verses OCDM), are justified by previous license amendments (e.g., surveillance requirements for T_{Hot} wide versus narrow range instrumentation), or correct logic errors (e.g., ECCS subsystem TS headings based on operating mode, with Mode 3 breakpoints based on pressurizer pressure and not temperature). The overly restrictive emergency power requirements for non critical single train quality related radiation monitors are being removed, while critical radiation monitor emergency power requirements are unaffected by the change. Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

Based on the above discussion and the supporting evaluation of Technical Specification changes, FPL has determined that the proposed license amendments involve no significant hazards consideration.

Environmental Consideration

The proposed license amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The proposed amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite, and no significant increase in individual or cumulative occupational radiation exposure. FPL has concluded that the proposed amendments involve no significant hazards consideration and meet the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and that, pursuant to 10 CFR 51.22(b), an environmental impact statement or environmental assessment need not be prepared in connection with issuance of the amendments.

ST. LUCIE UNIT 1 MARKED UP TECHNICAL SPECIFICATION PAGES

Updated per Amendment 170 dated 01/19/01

(4) Pursuant to the Act, and 10 CFR Parts 30, 40, and 70, to receive, possess and use in amounts as required any byprod ıct source or special nuclear material without restriction to chemical o· physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components;

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear m iterials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Sections 30.34 of Part 30, Section 40.41 of Part 40, Section 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commissior now or hereafter in effect; and is subject to the additional conditions specified or incorporated below;

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2700 megawatts (thermal), provided that the construction items, preoperational tests, sta tup tests, and other items identified in Enclosure 1 to this license have been completed as specified in Enclosure 1. Enclosure 1 is an integral part of, and is hereby incorporated in this license. [Amendment 48]

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 170 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications. [Amendment 170]

(3) Fire Protection

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The licensee shall implement and maintain in effect all provisions of the approved fire protection program as described in the Updated Final Safety Analysis Report for the facility (The fire protection program and features were originally described in licensee submittals L-83-514 dated October 7, 1983, L-83-227 dated April ~~22~~, 1983, L-83-261 dated April 25, 1983, L-83-453 dated August 24, 1983, L-83-488 dated September 16, 1983, L-83-588 dated December 14, 1983, L-84-346 dated November 28, 1984, L-84-390 dated December 31, 1984, and L-85-71 dated February 21, 1985) and as approved in by NRC letter dated July 17, 1984 and supplemented by NRC letters dated

Updated per Amendment 170 dated 01/19/01

February 21, 1985, March 5, 1987, and October 4, ~~1998~~ 1988 subject to the following provision:

> The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. [Amendment 115]

(4) DELETED PER AMENDMENT #88, DATED JANUARY 13, 1988.

(5) DELETED PER AMENDMENT #88, DATED JANUARY 13, 1988.

(6) SUSTAINED CORE UNCOVERY ACTIONS

Procedural guidance shall be in place to instruct operators to implement actions which are designed to mitigate a small break loss of coolant accident prior to a calculated time of sustained core uncovery. [Amendment No. 151]

D. This licensee shall fully implement and maintain in effect all provisions of the Commission-approved physical security, guard training and qualification, and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "St. Lucie Plant Security Plan," with revisions submitted through April 11, 1988; "St. Lucie Plant Training and Qualification Plan", with revisions submitted through August 8, 1985; and "St. Lucie Plant Safeguards Contingency Plan," with revisions through December 8, 1986. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein. [Amendment 97]

E. DELETED PER AMENDMENT #46, DATED NOVEMBER 3, 1981.

F. The licensee shall implement the following requirements for the protection of the environment:

(1) DELETED PER AMENDMENT #162, DATED JULY 2, 1999.

(2) DELETED PER AMENDMENT #50, DATED MAY 21 1982.

(3) DELETED PER AMENDMENT #39, DATED MARCH 10, 1981.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
6. LOSS OF POWER		
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	≥2900 volts with a 1 ± .5 second time delay	≥2900 volts with a 1 ± .5 second time delay
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	≥3831 volts with a 18 ± 2 second time delay	≥3831 volts with a 18 ± 2 second time delay
c. 480 volts Emergency Bus Undervoltage (Degraded Voltage)	≥415 volts with a ≤9 second time delay	≥415 volts with a ≤9 second time delay
7. AUXILIARY FEEDWATER (AFAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. SG 1A & 1B Level Low	≥ 19.0%	≥18.0%
8. AUXILIARY FEEDWATER ISOLATION		
a. Steam Generator ΔP-High	≤275 psid	89.2 to 281 psid
b. Feedwater Header High ΔP	≤150.0 psid	56.0 to 157.5 psid

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INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring instrumentation channels shown in Table 3.3-6 shall be OPERABLE~~*~~ with their alarm setpoints within the specified limits.

delete asterisk

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION shown in Table 3.3-6.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the modes and at the frequencies shown in Table 4.3-3.

~~*The emergency power source may be inoperable in Modes 5 or 6.~~

delete

TABLE 3.3-11

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION	
1. Pressurizer Water Level	~~3~~ 2	1	1, 6	|
2. Auxiliary Feedwater Flow Rate	1/pump	1/pump	7	|
3. RCS Subcooling Margin Monitor	2	1	1, 6	|
4. PORV Position Indicator Acoustic Flow Monitor	1/valve	1/valve	2	
5. PORV Block Valve Position Indicator	1/valve	1/valve	2	
6. Safety Valve Position Indicator	1/valve	1/valve	3	
7. Incore thermocouples	4/core quadrant	2/core quadrant	1, 6	|
8. Containment Sump Water Level (Narrow Range)	1*	1*	4, 5	
9. Containment Sump Water Level (Wide Range)	2	1	4, 5	
10. Reactor Vessel Level Monitoring System	2**	1**	4, 5	
11. Containment Pressure	2	1	1, 6	|

* The non-safety grade containment sump water level instrument may be substituted.

** Definition of OPERABLE: A channel is composed of eight (8) sensors in a probe, of which four (4) sensors must be OPERABLE.

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Attachment 3
Page 6 of 17

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS ~~(T_{avg} ≥ 325 °F~~ OPERATING)

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high-pressure safety injection (HPSI) pump,

b. One OPERABLE low-pressure safety injection pump, and

c. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal.

APPLICABILITY: MODES 1, 2 and 3*.

ACTION:

a. 1. With one ECCS subsystem inoperable only because its associated LPSI train is inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

2. With one ECCS subsystem inoperable for reasons other than condition a.1., restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

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* With pressurizer pressure ≥ 1750 psia.

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - ~~T_{avg} < 325°F~~ SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. In MODES 3* and 4#, one ECCS subsystem composed of one OPERABLE high pressure safety injection pump and one OPERABLE flow path capable of taking suction from the refueling water storage tank on a safety injection actuation signal and automatically transferring suction to the containment sump on a sump recirculation actuation signal.

b. Prior to decreasing the reactor coolant system temperature below 270°F a maximum of only one high pressure safety injection pump shall be OPERABLE with its associated header stop valve open.

c. Prior to decreasing the reactor coolant system temperature below 236°F all high pressure safety injection pumps shall be disabled and their associated header stop valves closed except as allowed by Specifications 3.1.2.1 and 3.1.2.3.

APPLICABILITY: MODES 3* and 4.
MODES 5 and 6 when the Pressurizer manway cover is in place and the reactor vessel head is on.

ACTION:

a. With no ECCS subsystems OPERABLE in MODES 3* and 4#, immediately restore one ECCS subsystem to OPERABLE status or be in COLD SHUTDOWN within 20 hours.

b. With RCS temperature below 270°F and with more than the allowed high pressure safety injection pump OPERABLE or injection valves and header isolation valves open, immediately disable the high pressure safety injection pump(s) or close the header isolation valves.

c. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.

4.5.3.2 The high pressure safety injection pumps shall be verified inoperable and the associated header stop valves closed prior to decreasing below the above specified Reactor Coolant System temperature and once per month when the Reactor Coolant System is at refueling temperatures.

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* With pressurizer pressure <1750 psia.

\# REACTOR COOLANT SYSTEM cold leg temperature above 250°F.

6.0 ADMINISTRATIVE CONTROLS

6.2 ORGANIZATION (continued)

UNIT STAFF

6.2.2 The unit organization shall be subject to the following:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Reactor Operator shall be in the control room when fuel is in the reactor. In addition, while the reactor is in MODE 1, 2, 3, or 4, at least one licensed Senior Reactor Operator shall be in the control room.

c. A health physics technician# shall be on site when fuel is in the reactor.

d. Either a licensed SRO or licensed SRO limited to fuel handling who has no concurrent responsibilities during this operation shall be present during fuel handling and shall directly supervise all CORE ALTERATIONS.

e. Deleted.

\# The health physics technician may be less than the minimum requirement for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

TWO UNITS WITH TWO SEPARATE CONTROL ROOMS

WITH UNIT 2 IN MODE 5 OR 6 OR DEFUELED		
POSITION	**NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION**	
	MODE 1, 2, 3 or 4	**MODE 5 or 6**
SS (SRO)	1[a]	1[a]
SRO	1	None
RO	2	1
AO	2	2[b]
STA *	1	None

WITH UNIT 2 IN MODE 1, 2, 3, or 4		
POSITION	**NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION**	
	MODE 1, 2, 3 or 4	**MODE 5 or 6**
SS (SRO)	1[a]	1[a]
SRO	1	None
RO	2	1
AO	2	1
STA *	1[c]	None

SS - Shift Supervisor with a Senior Reactor Operator's License on Unit 1
SRO - Individual with a Senior Reactor Operator's License on Unit 1
STA - Shift Technical Advisor
RO - Individual with a Reactor Operator's License on Unit 1
AO - Auxiliary Operator

Except for the Shift Supervisor, the Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 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 of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Supervisor from the Control Room while the unit is in MODE 1, 2, 3 or 4, an individual (other than the Shift Technical Advisor) with a valid SRO license shall be designated to assume the Control Room command function. During any absence of the Shift Supervisor from the Control Room while the unit is in MODE 5 or 6, an individual with a valid SRO or RO license shall be designated to assume the Control Room command function.

a/ Individual may fill the same position on Unit 2.

b/ One of the two required individuals may fill the same position on Unit 2.

c/ If STA position is filled by an STA qualified Shift Supervisor or dedicated STA, then the individual may fill the same position on Unit 2.

* A single, onsite STA position shall be manned in Mode 1, 2, 3, and 4 unless the Shift Supervisor meets the qualifications for the STA as required by Technical Specification 6.3.1 or an individual on each unit with a Senior Reactor Operator's license meets the qualifications for the STA as required by Technical Specification 6.3.1.

6.0 ADMINISTRATIVE CONTROLS

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the FRG Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in FRG activities at any one time.

MEETING FREQUENCY

6.5.1.4 The FRG shall meet at least once per calendar month and as convened by the FRG Chairman or his designated alternate.

QUORUM

6.5.1.5 The quorum of the FRG necessary for the performance of the FRG responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The Facility Review Group shall be responsible for:

a. Review of (1) all new procedures required by Specification 6.8 and all procedure changes that require a written 50.59 ~~safety~~ evaluation, (2) all programs required by Specification 6.8 and changes thereto, and (3) any other proposed procedures or changes thereto as determined by the Plant General Manager to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to Appendix A Technical Specifications.

d. Review of all proposed changes or modifications to unit systems or equipment that affect nuclear safety.

e. Investigation of all violations of the Technical Specifications, including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the President-Nuclear Division and to the Chairman of the Company Nuclear Review Board.

f. Review of all REPORTABLE EVENTS.

g. Review of unit operations to detect potential nuclear safety hazards.

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Plant General Manager or the Company Nuclear Review Board.

6.0 ADMINISTRATIVE CONTROLS

i. Not Used.

j. Not Used.

k. Review of every unplanned on-site release of radioactive material to the environs including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the President-Nuclear Division and to the Company Nuclear Review Board.

l. Review of changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL and RADWASTE TREATMENT SYSTEMS.

m. Review and documentation of judgment concerning prolonged operation in bypass, channel trip, and/or repair of defective protection channels of process variables placed in bypass since the last FRG meeting.

n. Review of the Fire Protection Program and implementing procedures and submittal of recommended changes to the Company Nuclear Review Board.

AUTHORITY

6.5.1.7 The Facility Review Group shall:

a. Recommend in writing to the Plant General Manager, approval or disapproval of items considered under Specifications 6.5.1.6.a through d above.

b. Render determinations in writing with regard to whether or not each item considered under Specifications 6.5.1.6.a, b, d, and e above ~~constitutes an unreviewed safety question~~. requires NRC approval pursuant to 10 CFR 50.59

c. Provide written notification within 24 hours to the President-Nuclear Division and the Company Nuclear Review Board of disagreement between the FRG and the Plant General Manager; however, the Plant General Manager shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1 above.

RECORDS

6.5.1.8 The Facility Review Group shall maintain written minutes of each FRG meeting that, at a minimum, document the results of all FRG activities performed under the responsibility and authority provisions of these Technical Specifications. Copies shall be provided to the Plant General Manager, President-Nuclear Division and the Chairman of the Company Nuclear Review Board.

ADMINISTRATIVE CONTROLS

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the CNRB Chairman to provide expert advice to the CNRB.

MEETING FREQUENCY

6.5.2.5 The CNRB shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per 6 months thereafter and as convened by the CNRB Chairman or his designated alternate.

QUORUM

6.5.2.5 The quorum of the CNRB necessary for the performance of the CNRB review and audit functions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least a majority of CNRB members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the facility.

REVIEW

6.5.2.7 The CNRB shall review:

a. The ~~safety~~ evaluations for (1) changes to procedures, equipment, or systems and (2) tests or experiments completed under the provisions ~~of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question~~. require NRC approval pursuant to 10 CFR 50.59.

b. Proposed changes to procedures, equipment, or systems which ~~involve an unreviewed safety question as defined in Section 50.59, 10 CFR~~. require NRC approval pursuant to 10 CFR 50.59

c. Proposed tests or experiments which ~~involve an unreviewed safety question as defined in Section 50.59, 10 CFR.~~ require NRC approval pursuant to 10 CFR 50.59

d. Proposed changes to Technical Specifications or this Operating License.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety.

6.0 ADMINISTRATIVE CONTROLS

6.8.3 CHANGES TO PROCEDURES

a. Each revision to the procedures of Specification 6.8.1a. through i. above shall be independently reviewed by an individual or group from the appropriate discipline(s), and revisions that require a written ~~safety~~ evaluation pursuant to 10 CFR 50.59 shall be reviewed by the FRG. Procedure revisions shall be approved by the Plant General Manager or individuals designated in writing by the Plant General Manager prior to implementation.

b. Temporary changes to procedures of Specification 6.8.1a. through i. above may be made provided:

1. The intent of the original procedure is not altered.

2. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

3. The change is documented and, if appropriate, incorporated in the next revision of the affected procedure pursuant to Specification 6.8.3.a.

6.8.4 The following programs shall be established, implemented, maintained, and shall be audited under the cognizance of the CNRB:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the Shutdown Cooling System, High Pressure Safety Injection System, Containment Spray System, and RCS Sampling. The program shall include the following:

(i) Preventive maintenance and periodic visual inspection requirements, and

(ii) Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radioiodine Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

(i) Training of personnel,

(ii) Procedures for monitoring, and

(iii) Provisions for maintenance of sampling and analysis equipment.

ADMINISTRATIVE CONTROLS

The provisions of T.S. 4.0.2 do not apply to test frequencies in the Containment Leak Rate Testing Program.

The provisions of T.S. 4.0.3 are applicable to the Containment Leak Rate Testing Program.

i. Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2 and 3 components (pumps and valves). The program shall include the following:

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

b. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice testing activities;

c. The provisions of Specification 4.0.3 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 technical specification.

* Where ASME Boiler and Pressure Vessel Code is referenced it also refers to the applicable portions of ASME/ANSI OM-Code, "Operation and Maintenance of Nuclear Power Plants," with applicable addenda, to the extent it is referenced in the Code.

Insert

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

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

STARTUP REPORT

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

Insert for Page 6-15c

j. Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

1. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

2. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

    a. a change in the TS incorporated in the license; or

    b. a change to the updated UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59

3. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

4. Proposed changes that meet the criteria of Specification 6.8.4.j.2.a or 6.8.4.j.2.b, above, shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

6.13 PROCESS CONTROL PROGRAM (PCP)

Changes to the PCP:

1. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.2q. This documentation shall contain:

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

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

2. Shall become effective after review and acceptance by the Facility Review Group and the approval of the Plant General Manager.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Changes to the ~~OCDM:~~ ODCM

1. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.2q. This documentation shall contain:

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

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

2. Shall become effective after review and acceptance by the Facility Review Group and the approval of the Plant General Manager.

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

ST. LUCIE UNIT 2 MARKED UP TECHNICAL SPECIFICATION PAGES

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure	M	R
2. Reactor Coolant Outlet Temperature - T_{Hot} (~~Narrow~~ Wide Range)	M	R
3. Reactor Coolant Inlet Temperature - T_{Cold} (Wide Range)	M	R
4. Reactor Coolant Pressure - Wide Range	M	R
5. Pressurizer Water Level	M	R
6. Steam Generator Pressure	M	R
7. Steam Generator Water Level - Narrow Range	M	R
8. Steam Generator Water Level - Wide Range	M	R
9. Refueling Water Storage Tank Water Level	M	R
10. Auxiliary Feedwater Flow Rate (Each pump)	M	R
11. Reactor Coolant System Subcooling Margin Monitor	M	R
12. PORV Position/Flow Indicator	M	R
13. PORV Block Valve Position Indicator	M	R
14. Safety Valve Position/Flow Indicator	M	R
15. Containment Sump Water Level (Narrow Range)	M	R
16. Containment Water Level (Wide Range)	M	R
17. Incore Thermocouples	M	R
18. Reactor Vessel Level Monitoring System	M	R

EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 ECCS SUBSYSTEMS - ~~T_{avg} GREATER THAN OR EQUAL TO 325°F~~ OPERATING

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE high pressure safety injection pump,

b. One OPERABLE low pressure safety injection pump, and

c. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal, and

d. One OPERABLE charging pump.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. 1. With one ECCS subsystem inoperable only because its associated LPSI train is inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

2. With one ECCS subsystem inoperable for reasons other than condition a.1., restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

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

* With pressurizer pressure greater than or equal to 1750 psia.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.3 ECCS SUBSYSTEMS - ~~T_{avg} LESS THAN 325°F~~ SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE high-pressure safety injection pump, and

b. An OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Sump Recirculation Actuation Signal.

APPLICABILITY: MODES 3* and 4*.
Footnote # shall remain applicable in MODES 5 and 6 when the Pressurizer manway cover is in place and the reactor vessel head is on.

ACTION:

a. With no ECCS subsystem OPERABLE, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

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

SURVEILLANCE REQUIREMENTS

4.5.3 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.

* With pressurizer pressure less than 1750 psia.
\# One HPSI shall be rendered inoperable prior to entering MODE 5.

ADMINISTRATIVE CONTROLS

MEETING FREQUENCY

6.5.1.4 The FRG shall meet at least once per calendar month and as convened by the FRG Chairman or his designated alternate.

QUORUM

6.5.1.5 The quorum of the FRG necessary for the performance of the FRG responsibility and authority provisions of these Technical Specifications shall consist of the Chairman or his designated alternate and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The Facility Review Group shall be responsible for:

a. Review of (1) all new procedures required by Specification 6.8 and all procedure changes that require a written ~~safety~~ 50.59 evaluation, (2) all programs required by Specification 6.8 and changes thereto, and (3) any other proposed procedures or changes thereto as determined by the Plant General Manager to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to Appendix A Technical Specifications.

d. Review of all proposed changes or modifications to unit systems or equipment that affect nuclear safety.

e. Investigation of all violations of the Technical Specifications, including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the President-Nuclear Division and to the Chairman of the Company Nuclear Review Board.

f. Review of all REPORTABLE EVENTS.

g. Review of unit operations to detect potential nuclear safety hazards.

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Plant General Manager or the Company Nuclear Review Board.

i. Not Used.

j. Not Used.

RESPONSIBILITIES (Continued)

k. Review of every unplanned on-site release of radioactive material to the environs including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the President-Nuclear Division and to the Company Nuclear Review Board.

l. Review of changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL and RADWASTE TREATMENT SYSTEMS.

m. Review and documentation of judgment concerning prolonged operation in bypass, channel trip, and/or repair of defective protection channels of process variables placed in bypass since the last FRG meeting.

n. Review of the Fire Protection Program and implementing procedures and submittal of recommended changes to the Company Nuclear Review Board.

AUTHORITY

6.5.1.7 The Facility Review Group shall:

a. Recommend in writing to the Plant General Manager approval or disapproval of items considered under Specifications 6.5.1.6a. through d. and m. above.

b. Render determinations in writing with regard to whether or not each item considered under Specifications 6.5.1.6a. b. d and e above ~~constitutes an unreviewed safety question~~. requires NRC approval pursuant to 10 CFR 50.59

c. Provide written notification within 24 hours to the President-Nuclear Division and the Company Nuclear Review Board of disagreement between the FRG and the Plant General Manager; however, the Plant General Manager shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1 above.

RECORDS

6.5.1.8 The Facility Review Group shall maintain written minutes of each FRG meeting that, at a minimum, document the results of all FRG activities performed under the responsibility and authority provisions of these technical specifications. Copies shall be provided to the Plant General Manager, President-Nuclear Division and the Chairman of the Company Nuclear Review Board.

6.5.2 COMPANY NUCLEAR REVIEW BOARD (CNRB)

FUNCTION

6.5.2.1 The Company Nuclear Review Board shall function to provide independent review and audit of designated activities in the areas of:

a. nuclear power plant operations
b. nuclear engineering
c. chemistry and radiochemistry
d. metallurgy

REVIEW

6.5.2.7 The CNRB shall review:

a. The ~~safety~~ evaluations for (1) changes to procedures, equipment, or systems and (2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not ~~constitute an unreviewed safety question~~. require NRC approval pursuant to 10CFR 50.59

b. Proposed changes to procedures, equipment, or systems which ~~involve an unreviewed safety question as defined in Section 50.59, 10 CFR.~~ require NRC approval pursuant to 10 CFR 50.59

c. Proposed tests or experiments which ~~involve an unreviewed safety question as defined in Section 50.59, 10 CFR~~. require NRC approval pursuant to 10 CFR 50.59

d. Proposed changes to Technical Specifications or this Operating License.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety.

g. All REPORTABLE EVENTS.

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety.

i. Reports and meetings minutes of the Facility Review Group.

AUDITS

6.5.2.8 Audits of unit activities shall be performed under the cognizance of the CNRB. These audits shall encompass:

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions.

b. The performance, training and qualifications of the entire unit staff.

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems, or method of operation that affect nuclear safety.

ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

f. Fire Protection Program implementation.

g. PROCESS CONTROL PROGRAM implementation.

h. OFFSITE DOSE CALCULATION MANUAL implementation.

i. Quality Control Program for effluent monitoring, using the guidance in Regulatory Guide 1.21, Revision 1, June 1974.

j. Quality Control Program for environmental monitoring using the guidance in Regulatory Guide 4.1, Revision 1, April 1975.

6.8.2 Review and Approval of Procedures:

Each new procedure of Specification 6.8.1a. through i. above shall be independently reviewed by an individual or group from the appropriate discipline(s), and shall be reviewed by the FRG. New procedures shall be approved by the Plant General Manager or individuals designated in writing by the Plant General Manager prior to implementation. Each procedure of Specification 6.8.1 shall be reviewed periodically as set forth in administrative procedures.

6.8.3 Changes to Procedures:

a. Each revision to the procedures of Specification 6.8.1a. through i. above shall be independently reviewed by an individual or group from the appropriate discipline(s), and revisions that require a written safety evaluation pursuant to 10 CFR 50.59 shall be reviewed by the FRG. Procedure revisions shall be approved by the Plant General Manager or individuals designated in writing by the Plant General Manager prior to implementation.

b. Temporary changes to procedures of Specification 6.8.1.a through i. above may be made provided:

1. The intent of the original procedure is not altered.

2. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

3. The change is documented and, if appropriate, incorporated in the next revision of the affected procedure pursuant to Specification 6.8.3.a.

ADMINISTRATIVE CONTROLS (Continued)

Leakage rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and C tests, $\leq 0.75\ L_a$ for Type A tests, and $\leq 0.12\ L_a$ for secondary containment bypass leakage paths.

b. Air lock testing acceptance criteria are:

1) Overall air lock leakage is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

2) For each door seal, leakage rate is $< 0.01\ L_a$ when pressurized to $\geq P_a$.

The provisions of T.S. 4.0.2 do not apply to test frequencies in the Containment Leak Rate Testing Program.

The provisions for T.S. 4.0.3 are applicable to the Containment Leak Rate Testing Program.

i. Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2 and 3 components (pumps and valves). The program shall include the following:

a. Testing frequencies specified in Section XI of 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

b. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice testing activities.

c. The provisions of Specification 4.0.3 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 technical specification.

* Where ASME Boiler and Pressure Vessel Code is referenced it also refers to the applicable portions of ASME/ANSI OM-Code, "Operation and Maintenance of Nuclear Power Plants," with applicable addenda, to the extent it is referenced in the Code.

Insert

Insert for Page 6-15c

j. Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications

1. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

2. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   a. a change in the TS incorporated in the license; or

   b. a change to the updated UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59

3 The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

4. Proposed changes that meet the criteria of Specification 6.8.4.j.2.a or 6.8.4.j.2.b, above, shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).