FPL

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

December 14, 2009

L-2009-262
10 CFR 50.90

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

Re: St. Lucie Units 1 and 2
Docket Nos. 50-335 and 50-389
Proposed License Amendment
Application to Delete Structural Integrity Technical Specifications, Update Accident Monitoring Instrumentation Requirements, and Minor Corrections

Pursuant to 10 CFR 50.90 and 10 CFR 50.91(a)(1), Florida Power and Light Company (FPL) requests approval of a change to St. Lucie Units 1 and 2 Facility Operating Licenses DPR-67 and NPF-16, respectively. Attached for Nuclear Regulatory Commission review and approval is a proposed Technical Specification (TS) change to remove the structural integrity requirements contained in TS 3/4.4.10 (Unit 1) and 3/4.4.11 (Unit 2) and their associated Bases from the St. Lucie TSs. Removal of the structural integrity TS is consistent with NUREG-1432 in that it does not meet the criteria of 10 CFR 50.36 for inclusion in the TSs. The proposed amendment will also incorporate changes to accident monitoring instrumentation for consistency with NUREG-1432 actions and allowed outage times for conditions that drive a unit to HOT SHUTDOWN. The proposed amendment also makes several administrative corrections based on obvious typos, previous amendments, or obsolete requirements.

The proposed change has been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c) and it has been determined that this change involves no significant hazards considerations. Attachment 1 provides an evaluation of the proposed change. Attachment 2 provides the existing TS pages marked up to show the proposed change. Attachment 3 provides the proposed TS changes in final typed format.

The proposed change is neither exigent nor emergency. Once approved, the amendment will be implemented within 60 days.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated State Official. If you should have any questions regarding this submittal, please contact Ken Frehafer at (772) 467-7748.

A001
NRR

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

Executed on the 14th day of December, 2009

Very truly yours,

Richard L. Anderson
Site Vice President
St. Lucie Plant

Attachments

cc: Mr. William Passetti, Florida Department of Health

Application to Delete Structural Integrity Technical Specifications, Update Accident Monitoring Instrumentation Requirements, and Minor Corrections

Analysis of Proposed Technical Specification Change

1.0 Description of Proposed Changes

1.1 Technical Specification (TS) 3/4.4.10 (Unit 1) and TS 3/4.4.11 (Unit 2)

The proposed change removes the St. Lucie structural integrity requirements contained in TSs 3/4.4.10 (Unit 1) and 3/4.4.11 (Unit 2) and the associated TS Bases from the TSs. The proposed change is consistent with NUREG-1432, Standard Technical Specifications Combustion Engineering Plants, Revision 3.0 (Reference 1).

1.2 Reactor Coolant Pump Flywheel Inspection (administrative in nature)

The change also relocates the Unit 2 reactor coolant pump (RCP) flywheel inspection requirements in Surveillance Requirement (SR) 4.4.11 to a new administrative TS program consistent with NUREG-1432.

1.3 Administrative TS 6.4.1 (administrative in nature)

The proposed change replaces the obsolete references to ANSI / ANS-3.1 - 1978 and 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees with the current NUREG-1432 Standard Technical Specifications Combustion Engineering Plants, Revision 3.0 (Reference 1) wording for Unit Staff Qualifications.

1.4 TS 3.3.3.8, Table 3.3-11 (Unit 1) and TS 3.3.3.6 (Unit 2)

The proposed changes to Unit 1 TS 3.3.3.8, Table 3.3-11 ACTION 1, 2, 6, and 7 and Unit 2 TS 3.3.3.6 ACTION 'a' and 'b' will revise the required end states and completion times.

1.5 Minor changes (administrative in nature)

- Unit 2 TS 3.1.2.6 – change the action to be logically consistent.
- Unit 2 TS Surveillance Requirement (SR) 4.3.3.2 – correct typo.
- Unit 1 TS index page VI – correct section heading for CONTAINMENT SYSTEMS
- Unit 2 TS index page XXIV – Table 3.6-1.
- Unit 2 TSs 6.8.4.l.2 and 6.9.1.13 pertain to inspection requirements for the St. Lucie Unit 2 original steam generators.

2.0 Proposed Change

2.1 TS 3/4.4.10 (Unit 1) and TS 3/4.4.11 (Unit 2)

TS Limiting Condition for Operation (LCO) 3/4.4.10 (Unit 1) and 3/4.4.11 (Unit 2), Structural Integrity, including its associated actions and SR 4.4.10 (Unit 1) and 4.4.11 (Unit 2) would be removed from the St. Lucie TSs and TS Bases.

2.2 RCP Flywheel Inspection

The RCP flywheel inspection requirements in Unit 2 SR 4.4.11 would be relocated to a new administrative TS program, 6.8.4.o. The prescribed inspection methods are unchanged and will be reworded to: "*Reactor Coolant Pump Flywheel Inspection Program - This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendation of Regulatory position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975.*" This wording is identical to the RCP Flywheel inspection program requirements in NUREG-1432, TS 5.5.7. Relocating the RCP flywheel inspection requirements to an administrative TS program will not revise any current requirements.

2.3 Administrative TS 6.4.1 Training Requirements

The existing text for TS 6.4.1 will be replaced by "*Each member of the unit staff shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, Revision 3.*"

2.4 TS 3.3.3.8, Table 3.3-11 (Unit 1) and TS 3.3.3.6 (Unit 2)

The proposed changes to Unit 1 TS 3.3.3.8, Table 3.3-11 ACTION 1, 2, 6, and 7 and Unit 2 TS 3.3.3.6 ACTION a and b will require the end states to be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

2.5 Minor changes (administrative in nature)

- Unit 2 TS 3.1.2.6 – change the action to be logically consistent.

  Change the ACTION to read "With no boric acid makeup pump required for the boron injection flow path(s) pursuant to Specification 3.1.2.2 *operable...*"

- Unit 2 TS SR 4.3.3.2 – correct typo.

Change the SR to read "At *least* once per 18 months..."

- Unit 1 TS index page VI – correct section heading for CONTAINMENT SYSTEMS

  Change the section heading to read "*3/4.6* CONTAINMENT SYSTEMS..."

- Unit 2 TS index page XXIV – Table 3.6-1.

  Delete Table 3.6.1.

- Unit 2 TSs 6.8.4.l.2 and 6.9.1.13

  Delete the TSs.

3.0 Background

3.1 TS 3/4.4.10 (Unit 1) and TS 3/4.4.11 (Unit 2)

The purpose of TSs 3/4.4.10 (Unit 1) and 3/4.4.11 (Unit 2), Structural Integrity, is to specify the requirements for maintaining the structural integrity of ASME Code Class 1, 2 and 3 components. This specification was originally intended to support assurance that structural integrity and operational readiness of these components are maintained at an acceptable level throughout the life of the facility. The specification is applicable in all operational modes. However, the specification does not provide actions for plant shutdown if its LCO is not met. This is because the specification addresses the passive pressure boundary function of American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Class 1, 2 and 3 components as established by compliance with the Inservice Inspection (ISI) program. The ISI program is required pursuant to 10 CFR 50.55a, Codes and Standards (Reference 2) and SR 4.0.5. This TS does not fulfill any of the criteria of 10 CFR 50.36(c)(2)(ii) (Reference 3) for retention in the TSs.

Maintaining a program-type requirement within an LCO creates significant interpretation issues for Operations personnel. The structural integrity TS was part of the original TSs and, therefore, no basis history is available regarding its intent. However, TSs 3/4.4.10 (Unit 1) and 3/4.4.11 (Unit 2) appear to have been included to help ensure that plant heatup and startup would not occur until all required portions of applicable systems were verified to meet ISI acceptance criteria following inspections performed during a plant outage (normally performed during refueling outages). Meeting this acceptance criteria helps ensure the integrity of applicable systems during all modes of operation, including accident events. For instance, the RCS pressure boundary is purposely breached during Mode 5 and 6 operations to support plant outage activities and such openings are not historically considered a violation of TS 3/4.4.10 (Unit 1) or TS 3/4.4.11 (Unit 2). Furthermore, TSs 3/4.4.10 (Unit 1) and 3/4.4.11 (Unit 2) contain no actions suggesting they were designed to accommodate integrity concerns once plant heatup has commenced. Structural integrity ISI activities are performed only during plant outages when conditions exist that permit access to the applicable systems and are not monitored or controlled through application of the ISI program during the operational cycle. For example, other TSs are designed to monitor the structural integrity of the RCS during operation and provide actions to shutdown the unit if compliance is not maintained. RCS heatup and cooldown rates (TSs 3.4.9.1 and 3.4.9.2), and the overpressure mitigation system (TS 3.4.9.3) protect against applying undue stresses as a result of pressure/temperature transients on RCS components and piping. RCS leakage TSs (3.4.6.1 and 3.4.6.2) provide a means of protecting the RCS integrity by detecting and monitoring leakage. Therefore, it is not necessary to apply TSs 3/4.4.10 (Unit 1) and 3/4.4.11 (Unit 2) when integrity issues become evident during plant operation above cold shutdown. Because TSs 3/4.4.10 (Unit 1) and 3/4.4.11 (Unit 2) are redundant to other regulations, it is acceptable to remove TS 3/4.4.10 (Unit 1) and TS 3/4.4.11 (Unit 2) requirements from the TSs.

Removing these specifications does not reduce the controls that are necessary to ensure compliance with the ASME Code. Structural integrity is maintained by compliance with 10 CFR 50.55a as implemented through the St. Lucie ISI program required by TS 4.0.5, as well as by compliance with TSs 3.4.6.1, 3.4.6.2, 3.4.9.1, 3.4.9.2 and 3.4.9.3 for the RCS.

3.2 RCP Flywheel Inspection

The Unit 2 RCP flywheel inspection requirements in SR 4.4.11 would be relocated to a new administrative TS program, 6.8.4.o. The prescribed inspection methods are unchanged. With the addition of the existing RCP flywheel inspection requirements as an administrative TS program consistent with NUREG-1432, no surveillance requirements will be revised as a result of the relocation.

3.3 Administrative TS 6.4.1 Training

The St. Lucie Units 1 and 2 UFSARs, Sections 13.2, describe the training program as meeting or exceeding the requirements and recommendations of Section 5.5 of ANSI/ANS-3.1 1978 and 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of enclosure 1 of the March 28, 1980 NRC letter to all licensees as outlined in Section 6.4, Training, of the plant TSs.

3.4 TS 3.3.3.8, Table 3.3-11 (Unit 1) and TS 3.3.3.6 (Unit 2)

The primary purpose of the post accident monitoring (PAM) instrumentation is to display plant variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take the manual actions, for which no automatic control is provided, that are required for safety systems to accomplish their safety functions for design basis events.

3.5 Minor changes (administrative in nature) – the justification for the following administrative issues will be discussed in the next section.

- Unit 2 TS 3.1.2.6
- Unit 2 TS SR 4.3.3.2 – correct typo.
- Unit 1 TS index page VI – correct section heading for CONTAINMENT SYSTEMS
- Unit 2 TS index page XXIV – Table 3.6-1.
- Unit 2 TSs 6.8.4.l.2 and 6.9.1.13

4.0 Regulatory Analysis

4.1 TS 3/4.4.10 (Unit 1) and TS 3/4.4.11 (Unit 2)

Section 182a of the Atomic Energy Act, as amended (the Act), requires applicants for nuclear power plant operating licenses to incorporate TSs as part of the license. The Commission's regulatory requirements related to the content of the TSs are set forth in Title 10 of the Code of Federal Regulations (10 CFR) 50.36. That regulation requires that the TSs include items in five categories, including: (1) safety limits, limiting safety system settings and limiting control settings, (2) limiting conditions for operation, (3) surveillance requirements, (4) design features, and (5) administrative controls.

On July 22, 1993, the Commission issued its Final Policy Statement, expressing the view that satisfying the guidance in the policy statement also satisfies Section 182a of the Act and 10 CFR 50.36. The Final Policy Statement gave guidance for evaluating the required scope of the TSs and defined the guidance criteria to be used in determining which of the LCOs and associated SRs should remain in the TSs. The Commission noted that, in allowing certain items to be relocated to licensee-controlled documents while requiring that other items be retained in the TSs, it was adopting the qualitative standard enunciated by the Atomic Safety and Licensing Appeal Board in Portland General Electric Co. (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). There, the Appeal Board observed:

[T]here is neither a statutory nor a regulatory requirement that every operational detail set forth in an applicant's safety analysis report (or equivalent) be subject to a technical specification, to be included in the license as an absolute condition of operation which is legally binding upon the licensee unless and until changed with specific Commission approval. Rather, as best we can discern it, the contemplation of both the Act and the regulations is that technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety.

By this approach, existing LCO requirements that fall within or satisfy any of the criteria in the Final Policy Statement should be retained in the TSs; those LCO requirements that do not fall within or satisfy these criteria may be relocated to licensee-controlled documents.

The Commission codified the four criteria in 10 CFR 50.36 (60 FR 36953, July 19, 1995). The four criteria are stated as follows:

(1) Installed instrumentation that is used to detect, and indicate in a control room, a significant abnormal degradation of the reactor coolant pressure boundary;

(2) A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;

(3) A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier; and

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

As a result, existing LCO requirements that fall within or satisfy any of the criteria in 10 CFR 50.36(c)(2)(ii) must be retained in the TSs while those LCO requirements that do not fall within or satisfy these criteria may be relocated to other licensee-controlled documents.

4.2 RCP Flywheel Inspection

The Unit 2 RCP flywheel inspection requirements in SR 4.4.11 would be relocated to a new administrative TS program, 6.8.4.o. The prescribed inspection methods are unchanged. With the addition of the existing RCP flywheel inspection requirements as an administrative TS program consistent with NUREG-1432, no surveillance requirements will be revised as a result of the relocation. Therefore, this change is administrative in nature and will not be evaluated further in this amendment request.

4.3 Administrative TS 6.4.1 Training

St. Lucie plant training programs are accredited through the National Nuclear Accrediting Board (NNAB) and have been for over 20 years. The NRC has agreed that this an acceptable alternative to some requirements outlined in 10 CFR 50.120 (regarding training programs) and 10 CFR 55 (regarding operator licensing) where a "Systems Approach to Training (SAT)" is used in lieu of the stated CFR requirements.

On March 19, 1987, Generic Letter (GL) 87-07, "Information Transmittal of Final Rulemaking for Revisions to Operator Licensing - 10 CFR Part 55 and Conforming Amendments," informed facility licensees that they had the option of substituting an accredited, SAT-based program for their operator training program previously approved by the NRC. The GL indicated that this option may be implemented upon written notification to the NRC and that it did not require any staff review. The GL also noted the NRC's expectation that facility licensees would update their licensing basis documents (e.g., their final safety analysis reports (FSARs) and technical specifications (TSs)), as necessary, to conform with their accredited program status.

As stated in RIS 2001-001, the NRC has not changed its requirements or position with regard to license eligibility for senior reactor operators and reactor operators since 1987. Regulatory Guide (RG) 1.8 (Revision 2 or 3) and the NANT's guidelines for education and experience (those that were in effect in 1987 or those that were issued in January 2000) outline acceptable methods for implementing the Commission's regulations in this area. As stated in the RIS, any required TS changes would be considered administrative in nature.

FPL chose to replace the existing administrative TS training wording with the NUREG-1432 wording for training.

Therefore, this change is administrative in nature and will not be evaluated further in this amendment request.

4.4 TS 3.3.3.8, Table 3.3-11 (Unit 1) and TS 3.3.3.6 (Unit 2)

St. Lucie has custom TS, and when an LCO is not met and the required action allowed outage time expires the action statements normally step through all intervening modes to get into a condition where the LCO is not applicable. However, the St. Lucie accident monitoring instrumentation TSs do not follow this standard. The St. Lucie accident monitoring TS LCOs are applicable in Modes 1, 2, and 3. However, the action statements either drive the end state to Mode 3, HOT STANDBY, or they drive the end state to Mode 4, HOT SHUTDOWN, with no intervening step through Mode 3, HOT STANDBY. The NUREG 1432, Rev. 3, Standard Technical Specifications Combustion Engineering Plants, Revision 3.0, completion times for post accident monitoring instrumentation (analog) contain a structured way to transition to Mode 4, HOT SHUTDOWN, when the LCO actions and completion times are not satisfied for instrumentation. This TS change will follow the conventions of NUREG-1432 for St. Lucie accident monitoring instrumentation whose actions should drive the unit to Mode 4, HOT SHUTDOWN conditions.

4.5 Minor corrections

- Unit 2 TS 3.1.2.6 – This change corrects an obvious error in the logic of the action statement.
- Unit 2 TS SR 4.3.3.2 – This change corrects an obvious typo in the action statement.

- Unit 1 TS index page VI – This change corrects an obvious typo in the TS index section heading for CONTAINMENT SYSTEMS.
- Unit 2 TS index page XXIV – This change deletes Table 3.6-1 from the TS index. This table was removed from the TS in Amendment 88 that implemented 10 CFR 50 Appendix J, Option B (TAC Nos. M97156/M97157).
- Unit 2 TSs 6.8.4.l.2 and 6.9.1.13 – These TSs pertain to the steam generator integrity program and reporting requirements for the St. Lucie Unit 2 original steam generators and are no longer applicable to the replacement steam generators that were installed in the SL2-17 refueling outage. The replacement steam generator inspection/reporting requirements are unchanged and contained in TSs 6.8.4.l.1 and 6.9.1.12.

Because these changes are administrative in nature, they will not be evaluated further in this amendment request.

5.0 Technical Analysis

5.1 TS 3/4.4.10 (Unit 1) and TS 3/4.4.11 (Unit 2)

The purpose of TS 3/4.4.10 (Unit 1) and TS 3/4.4.11 (Unit 2), Structural Integrity, is to specify the requirements of maintaining the structural integrity of ASME Code Class 1, 2 and 3 components. However, this is redundant to and does not contain the detail of the requirements contained within 10 CFR 50.55a. 10 CFR 50.36(c)(2)(ii) states that a TS LCO of a nuclear reactor must be established for each item meeting one or more of the following criteria:

Criterion 1 Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

TS 3/4.4.10 (Unit 1) and 3/4.4.11 (Unit 2) is not applicable to installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the RCS. Structural Integrity TS 3/4.4.10 (Unit 1) and 3/4.4.11 (Unit 2) does not meet Criterion 1.

Criterion 2 A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

TS 3/4.4.10 (Unit 1) and 3/4.4.11 (Unit 2) is not applicable to a process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. Although the specification is related to the integrity of applicable systems, compliance is maintained by meeting the requirements of 10 CFR 50.55a through implementation of the St. Lucie ISI program required by TS 4.0.5 and is not specifically monitored or controlled during plant operation. Structural Integrity TS 3/4.4.10 (Unit 1) and 3/4.4.11 (Unit 2) does not meet Criterion 2.

Criterion 3 A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

No specific TS-related structure, system, or component (SSC) is being revised or removed from the TSs. Each TS SSC must continue to meet the requirements of 10 CFR 50.55a as implemented through the St. Lucie ISI program required by TS 4.0.5. Structural Integrity TS 3/4.4.10 (Unit 1) and 3/4.4.11 (Unit 2) does not meet Criterion 3.

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

As stated above, no specific TS-related structure, system, or component (SSC) is being revised or removed from the TSs. Each TS SSC must continue to meet the requirements of 10 CFR 50.55a as implemented through the St. Lucie ISI program required by TS 4.0.5. Structural Integrity TS 3/4.4.10 (Unit 1) and 3/4.4.11 (Unit 2) does not meet Criterion 4.

The scope of this specification has been evaluated against the criteria of 10 CFR 50.36(c)(2)(ii) and none of these criteria require that the structural integrity controls specified in TS 3/4.4.10 (Unit 1) and 3/4.4.11 (Unit 2) are appropriate for retention in the St. Lucie TSs. This conclusion is consistent with

NUREG-1432, Standard Technical Specifications Combustion Engineering Plants, Revision 3.0.

Based on the above discussion, removal of structural integrity requirements contained in TS 3/4.4.10 (Unit 1) and 3/4.4.11 (Unit 2) from the TSs is acceptable.

5.2 Unit 2 RCP Flywheel Inspection – This is an administrative change.

5.3 Administrative TS 6.4.1 – This is an administrative change.

5.4 TS 3.3.3.8, Table 3.3-11 (Unit 1) and TS 3.3.3.6 (Unit 2)

The St. Lucie Unit 1 TS 3.3.3.8 Table 3.3-11 Actions 1 and 2 have a specified end state of HOT STANDBY. The required end state where the LCO is no longer applicable needs to be HOT SHUTDOWN. Consistent with NUREG-1432 TSs for post accident monitoring instrumentation, the proposed change drives the action end state to HOT SHUTDOWN (with a completion time of 12 hours) through the intervening state of HOT STANDBY (with a completion time of 6 hours). The proposed HOT STANDBY completion times are equal to or more conservative than the existing MODE 3 completion times and are acceptable. The HOT SHUTDOWN end state and completion time are acceptable as they are consistent with NUREG-1432 requirements.

St. Lucie Unit 1 TS 3.3.3.8, Table 3.3-11 Actions 6 and 7, and St. Lucie TS 3.3.3.6 actions 'a' and 'b' have a specified end state of HOT SHUTDOWN with a completion time of 12 hours. The allowed outage time, 12 hours, is consistent with the NUREG-1432 allowed outage time for entry into Mode 4 and is acceptable. FPL proposes to include the 6 hour allowed outage time to enter the intervening state of HOT STANDBY. This change is consistent with the NUREG-1432 completion time for entry into Mode 3 conditions.

Based on the above, there is no increase with any accident mitigation risk associated with the change. The proposed allowed outage times and the intervening step through HOT STANDBY are consistent with the equivalent NUREG-1432 completion times and actions for post accident instrumentation and are equal to or more conservative than the current TS requirements.

5.5 Minor changes – These changes are all administrative.

6.0 Determination of No Significant Hazards Consideration

FPL is proposing that the St. Lucie Operating Licenses be amended to revise the TS requirements for structural integrity, accident monitoring instrumentation, and make several administrative corrections based on obvious typos, previous amendments, or obsolete requirements. The proposed changes will remove Structural Integrity TSs 3/4.4.10 (Unit 1) and 3/4.4.11 (Unit 2) from the TSs. This specification is redundant to ASME Code compliance as required by 10 CFR 50.55a and specified in TS 4.0.5. Additionally, these proposed changes provide consistency for accident monitoring instrumentation actions and allowed outage times for conditions that drive the unit to HOT SHUTDOWN conditions. The proposed changes are consistent with NUREG-1432, Standard Technical Specifications Combustion Engineering Plants, Revision 3.0.

FPL has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, Issuance of Amendment, as discussed below:

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

   No. The proposed change to remove structural integrity controls from the TSs does not impact any mitigation equipment or the ability of the RCS pressure boundary to fulfill any required safety function. The proposed change will continue to ensure the requirements of 10 CFR 50.55a are maintained as specified in TS 4.0.5 and the new administrative TS program for RCP flywheel inspections. The changes to the accident instrumentation actions and allowed outage time have no appreciable effect on accident initiation or mitigation. Since no other accident mitigation or initiators are impacted by this change, no design basis accidents are affected.

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

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

   The proposed change will not alter the plant configuration or change the manner in which the plant is operated. Structural integrity will continue to be maintained as required by 10 CFR 50.55a and specified in TS 4.0.5 and the new administrative TS program for RCP flywheel inspections. Accident monitoring instrumentation does not contribute to failure modes. No new failure modes are being introduced by the proposed change.

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

3. Does the proposed change involve a significant reduction in the margin of safety?

   Removing TSs 3/4.4.10 (Unit 1) and 3/4.4.11 (Unit 2) from the TSs does not reduce the controls that are required to maintain the structural integrity of ASME Code Class 1, 2, or 3 components. There is no increase with any accident mitigation risk associated with the accident monitoring instrumentation TS changes as the proposed allowed outage times and the intervening step through HOT STANDBY are consistent with the equivalent NUREG-1432 completion times and actions for post accident instrumentation and are equal to or more conservative than the current TS requirements. No other safety margins are impacted due to the proposed change.

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

Based on the above, FPL concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of no significant hazards consideration is justified.

7.0 Environmental Considerations

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

8.0 Precedence

The proposed change to remove TS 3/4.4.10 (Unit 1) and 3/4.4.11 (Unit 2) from the TSs is consistent with NUREG-1432, Standard Technical Specifications Combustion Engineering Plants, Revision 3.0 and is similar to the amendment issued for Arkansas Nuclear One, Unit No. 2 in Amendment 270 dated March 1, 2007 (ML070570506) and the amendment request currently under review for Turkey Point Units 3 and 4 dated February 16, 2009 (ML090630238).

9.0 References

1. NUREG 1432, Standard Technical Specifications Combustion Engineering Plants, Revision 3.0

2. 10 CFR 50.55a, Codes and Standards

3. 10 CFR 50.36, Technical Specifications

Attachment 2

Application to Delete Structural Integrity Technical Specifications, Update Accident Monitoring Instrumentation Requirements, and Minor Corrections

Proposed Technical Specification Changes (mark-up)

Unit 1

Page V
Page VI
Page 3/4 3-43
Page 3/4 4-26
Page 6-6

Unit 2

Page VI
Page XXIV
Page 3/4 1-12
Page 3/4 3-24
Page 3/4 3-41
Page 3/4 4-39
Page 6-7
Page 6-15f
Page 6-15g
Page 6-15h
Page 6-15i
Page 6-15j
Page 6-20f

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

TABLE 3.3-11 (continued)

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels less than the Total No. of Channels shown in Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 30 days ~~or be in HOT STANDBY within the next 12 hours.~~

ACTION 2 - With position indication inoperable, restore the inoperable indicator to OPERABLE status or close the associated PORV block valve and remove power from its operator within 48 hours ~~or be in HOT STANDBY within the next 6 hours.~~

ACTION 3 - With any individual valve position indicator inoperable, obtain quench tank temperature, level and pressure information once per shift to determine valve position.

ACTION 4 - With the number of OPERABLE Channels one less than the Total Number of Channels shown in Table 3.3-11, either restore the inoperable channel to OPERABLE status within 7 days if repairs are feasible without shutting down or prepare and submit a Special Report to the Commission pursuant to the specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

ACTION 5 - With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

1. Initiate an alternate method of monitoring the reactor vessel inventory; and
2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status; and
3. Restore the Channel to OPERABLE status at the next scheduled refueling.

ACTION 6 - With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 48 hours ~~or be at least in HOT SHUTDOWN within the next 12 hours.~~

ACTION 7 - With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 72 hours ~~or be at least in HOT SHUTDOWN within the next 12 hours.~~

~~**REACTOR COOLANT SYSTEM**~~

~~**3.4.10 STRUCTURAL INTEGRITY**~~

~~**ASME CODE CLASS 1, 2, AND 3 COMPONENTS**~~

~~**LIMITING CONDITION FOR OPERATION**~~

~~3.4.10.1 The structural integrity of ASME Code Class 1, 2 and 3 components (except steam generator tubes) shall be maintained in accordance with Specification 4.4.10.1.~~

~~**APPLICABILITY**: All MODES.~~

~~**ACTION**:~~

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

~~b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.~~

~~c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.~~

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

~~**SURVEILLANCE REQUIREMENTS**~~

~~4.4.10.1 No additional Surveillance Requirements other than those required by Specification 4.0.5.~~

6.0 ADMINISTRATIVE CONTROLS

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI / ANS-3.1-1978 for comparable positions, except for:

(1) the radiation protection manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975,

(2) the Shift Technical Advisor who shall have specific training in plant design and plant operating characteristics, including transients and accidents, and any of the following educational requirements:

- Bachelor's degree in engineering from an accredited institution; or
- Professional Engineer's (PE) license obtained by successful completion of the PE examination; or
- Bachelor's degree in engineering technology from an accredited institution, including course work in the physical, mathematical, or engineering sciences, or
- Bachelor's degree in physical science from an accredited institution, including course work in the physical, mathematical, or engineering sciences.

(3) the Multi-Discipline Supervisors who shall meet or exceed the following requirements:

a. Education: Minimum of a high school diploma or equivalent.

b. Experience: Minimum of four years of related technical experience, which shall include three years power plant experience of which one year is at a nuclear power plant.

c. Training: Complete the Multi-Discipline Supervisor training program.

For the purpose of 10 CFR 55.4, a licensed senior reactor operator and a licensed reactor operator are those individuals who, in addition to meeting the requirements of 6.3.1, perform the functions described in 10 CFR 50.54(m).

6.4 TRAINING

6.4.1 ~~A retraining and replacement training program for the unit staff shall be maintained under the direction of the training manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI / ANS 3.1 – 1978 and 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.~~

Each member of the unit staff shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, Revision 3.

6.5 DELETED

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIST OF TABLES (Continued)

DELETED

REACTIVITY CONTROL SYSTEMS

BORIC ACID MAKEUP PUMPS – OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 At least the boric acid makeup pump(s) in the boron injection flow path(s) required OPERABLE pursuant to Specification 3.1.2.2 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if the flow path through the boric acid pump(s) in Specification 3.1.2.2 is OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

operable

With no boric acid makeup pump required for the boron injection flow path(s) pursuant to Specification 3.1.2.2 ~~inoperable~~, restore the boric acid makeup pump to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to its COLR limit at 200°F; restore the above required boric acid makeup pump(s) to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.6 The above required boric acid makeup pump(s) shall be demonstrated OPERABLE by verifying that the pump(s) develop the specified discharge pressure when tested pursuant to the Inservice Testing Program.

INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

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/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip 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 for the MODES and at the frequencies shown in Table 4.3-3.

4.3.3.2 At ~~lease~~ once per 18 months, each Control Room Isolation radiation monitoring instrumentation channel shall be demonstrated OPERABLE by verifying that the response time of the channel is within limits.

least

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours

a.* With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown in Table 3.3-10, either restore the inoperable channel to OPERABLE status within 7 days, ~~or be in HOT SHUTDOWN within the next 12 hours.~~

b.* With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-10, either restore the inoperable channels to OPERABLE status within 48 hours ~~or be in at least HOT SHUTDOWN within the next 12 hours.~~

c.** With the number of OPERABLE Channels one less than the Total Number of Channels shown in Table 3.3-10, either restore the inoperable channel to OPERABLE status within 7 days if repairs are feasible without shutting down or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

d.** With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirements of Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

1. Initiate an alternate method of monitoring the reactor vessel inventory; and

2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status, and

3. Restore the Channel to OPERABLE status at the next scheduled refueling.

e. The provisions of Specification 3.0.4 are not applicable.

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* Action statements do not apply to Reactor Vessel Level Monitoring System, Containment Sump Water Level (narrow range) and Containment Sump Water Level (wide range) instruments.

** Action statements apply only to Reactor Vessel Level Monitoring System, Containment Sump Water Level (narrow range) and Containment Sump Water Level (wide range) instruments.

~~**REACTOR COOLANT SYSTEM**~~

~~**3/4.4.11 STRUCTURAL INTEGRITY**~~

~~**LIMITING CONDITION FOR OPERATION**~~

~~3.4.11 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Specification 4.4.11.~~

~~**APPLICABILITY:** ALL MODES~~

~~**ACTION:**~~

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

~~b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.~~

~~c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component from service.~~

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

~~**SURVEILLANCE REQUIREMENTS**~~

~~4.4.11 In addition to the requirements of Specification 4.0.5, each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.~~

6.0 ADMINISTRATIVE CONTROLS

6.4 TRAINING

6.4.1 ~~A retraining and replacement training program for the unit staff shall be maintained under the direction of the training manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI / ANS 3.1 1978 and 10 CFR Part 55 and the supplemental requirements specified in Section A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.~~

Each member of the unit staff shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, Revision 3.

6.5 DELETED

ADMINISTRATIVE CONTROLS (continued)

l. Steam Generator (SG) Program (continued)

1. (continued)

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outages nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary-to-secondary leakage

~~2. A SG Program shall be established and implemented for the original SGs to ensure that SG tube integrity is maintained. In addition, the SG Program shall include the following provisions:~~

~~a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during a SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging or repair of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected, plugged or repaired to confirm that the performance criteria are being met.~~

ADMINISTRATIVE CONTROLS (continued)

~~l.~~ ~~Steam Generator (SG) Program~~ ~~(continued)~~

~~2.~~ ~~(continued)~~

~~b.~~ ~~Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage.~~

~~1.~~ ~~Structural integrity performance criterion: All in-service SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.~~

~~2.~~ ~~Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 0.3 gallons per minute total through all SGs and 216 gallons per day through any one SG.~~

~~3.~~ ~~The operational leakage performance criterion is specified in LCO 3.4.6.2.c, "Reactor Coolant System Operational Leakage."~~

~~c.~~ ~~Provisions for SG tube repair criteria~~

~~1.~~ ~~Tubes found by inservice inspection to contain a flaw in a non-sleeved region with a depth equal to or exceeding 40 percent of the nominal tube wall thickness shall be plugged or repaired except if permitted to remain in service through application of the alternate tube repair criteria discussed in Technical Specification 6.8.4.l.2.c.4.~~

~~2.~~ ~~Tubes found by inservice inspection to contain a flaw in (a) a sleeve or (b) the pressure boundary portion of the original tube wall in the sleeve to tube joint shall be plugged.~~

~~3.~~ ~~All tubes with sleeves that have a nickel band shall be plugged after one cycle in operation.~~

~~4.~~ ~~The C* methodology, as described below, may be applied to the expanded portion of the tube in the hot-leg tubesheet region as an alternative to the 40-percent depth based criteria of Technical Specification 6.8.4.l.2.c.1.~~

ADMINISTRATIVE CONTROLS (continued)

~~l.~~ ~~Steam Generator (SG) Program (continued)~~

~~2.~~ ~~c.~~ ~~4.~~ ~~(continued)~~

~~i.~~ ~~Tubes with no portion of a lower sleeve joint in the hot-leg tubesheet region shall be repaired or plugged upon detection of any flaw identified within 10.3 inches below the bottom of the hot-leg expansion transition or top of the tubesheet, whichever elevation is lower. Flaws located below this elevation may remain in service regardless of size.~~

~~ii.~~ ~~Tubes which have any portion of a sleeve joint in the hot-leg tubesheet region shall be plugged upon detection of any flaw that is located below the lower sleeve to tube joint and within 10.3 inches below the bottom of the hot-leg expansion transition or top of the tubesheet, whichever elevation is lower.~~

~~d.~~ ~~Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. For tubes with no portion of a lower sleeve joint in the hot-leg tubesheet region, the portion of the tube below 10.3 inches from the top of the hot leg tubesheet or expansion transition, whichever is lower, is excluded when the alternate repair criteria in TS Section 6.8.4.1.2.c.4 are applied. In tubes repaired by sleeving, the portion of the original tube wall between the sleeve's joints is not an area requiring inspection. In addition to meeting the requirements of d.1, d.2, d.3 and d.4 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.~~

~~1.~~ ~~Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.~~

~~2.~~ ~~Inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one refueling outage (whichever is less) without being inspected.~~

~~3.~~ ~~Inspect 100 percent of all inservice sleeves and sleeve-to-tube joints every 24 effective full power months or one refueling outage (whichever is less).~~

~~4.~~ ~~If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.~~

ADMINISTRATIVE CONTROLS (continued)

~~l. Steam Generator (SG) Program (continued)~~

~~2. (continued)~~

~~e. Provisions for monitoring operational primary-to-secondary leakage.~~

~~f. Provisions for SG tube repair methods. Steam generator tube repair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of these Specifications, tube plugging is not a repair. All acceptable tube repair methods are listed below.~~

~~1. Westinghouse Leak Limiting Alloy 800 sleeves as described in WCAP-15918-P Revision 2 (with range of conditions as revised in Appendix A of WCAP-16489-NP, Revision 0). Leak Limiting Alloy 800 Sleeves are applicable only to the original steam generators. Prior to installation of each sleeve, the location where the sleeve joints are to be established shall be inspected.~~

m. Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Air Cleanup System (CREACS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident.

The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREACS, operating at the flow rate required by the VFTP, at a Frequency of 36 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 36 month assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 4.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

ADMINISTRATIVE CONTROLS (continued)

n. 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:

(i) 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 or a water and sediment content within limits;

(ii) Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and

(iii) Total particulate concentration of the fuel oil is $\leq$ 10 mg/l when tested every 31 days.

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

o. Reactor Coolant Pump Flywheel Inspection Program

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

~~STEAM GENERATOR TUBE INSPECTION REPORT (continued)~~

~~6.9.1.13~~ ~~A report shall be submitted within 180 days after the initial entry into HOT SHUTDOWN following completion of an inspection of the original SGs performed in accordance with Specification 6.8.4.l.2. The report shall include:~~

- ~~a. The scope of inspections performed on each SG,~~
- ~~b. Active degradation mechanisms found,~~
- ~~c. Nondestructive examination techniques utilized for each degradation mechanism,~~
- ~~d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,~~
- ~~e. Number of tubes plugged or repaired during the inspection outage for each active degradation mechanism,~~
- ~~f. Total number and percentage of tubes plugged or repaired to date,~~
- ~~g. The results of condition monitoring, including the results of tube pulls and in-situ testing,~~
- ~~h. The effective plugging percentage for all plugging and tube repairs in each SG, and~~
- ~~i. Repair method utilized and the number of tubes repaired by each repair method.~~

~~The following information concerning indications found in the tubesheet region (including the expansion transition) shall be included in this report:~~

- ~~j. Number of total indications, location of each indication, orientation of each indication, severity of each indication, and whether the indications initiated from the inside or outside diameter.~~
- ~~k. The cumulative number of indications detected in the tubesheet region as a function of elevation within the tubesheet.~~
- ~~l. Projected end-of-cycle accident induced leakage from tubesheet indications.~~

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC within the time period specified for each report.

6.10 DELETED

Application to Delete Structural Integrity Technical Specifications, Update Accident Monitoring Instrumentation Requirements, and Minor Corrections

Word Processed TS

Unit 1

Page V
Page VI
Page 3/4 3-43
Page 3/4 4-26
Page 6-6

Unit 2

Page VI
Page XXIV
Page 3/4 1-12
Page 3/4 3-24
Page 3/4 3-41
Page 3/4 4-39
Page 6-7
Page 6-15f
Page 6-15g
Page 6-15i
Page 6-15j
Page 6-20f

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

TABLE 3.3-11 (continued)

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels less than the Total No. of Channels shown in Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 30 days or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

ACTION 2 - With position indication inoperable, restore the inoperable indicator to OPERABLE status or close the associated PORV block valve and remove power from its operator within 48 hours or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

ACTION 3 - With any individual valve position indicator inoperable, obtain quench tank temperature, level and pressure information once per shift to determine valve position.

ACTION 4 - With the number of OPERABLE Channels one less than the Total Number of Channels shown in Table 3.3-11, either restore the inoperable channel to OPERABLE status within 7 days if repairs are feasible without shutting down or prepare and submit a Special Report to the Commission pursuant to the specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 5 - With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

1. Initiate an alternate method of monitoring the reactor vessel inventory; and

2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status; and

3. Restore the Channel to OPERABLE status at the next scheduled refueling.

ACTION 6 - With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

ACTION 7 - With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 72 hours or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

DELETED

6.0 ADMINISTRATIVE CONTROLS

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI / ANS-3.1-1978 for comparable positions, except for:

(1) the radiation protection manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975,

(2) the Shift Technical Advisor who shall have specific training in plant design and plant operating characteristics, including transients and accidents, and any of the following educational requirements:

- Bachelor's degree in engineering from an accredited institution; or
- Professional Engineer's (PE) license obtained by successful completion of the PE examination; or
- Bachelor's degree in engineering technology from an accredited institution, including course work in the physical, mathematical, or engineering sciences, or
- Bachelor's degree in physical science from an accredited institution, including course work in the physical, mathematical, or engineering sciences.

(3) the Multi-Discipline Supervisors who shall meet or exceed the following requirements:

a. Education: Minimum of a high school diploma or equivalent.

b. Experience: Minimum of four years of related technical experience, which shall include three years power plant experience of which one year is at a nuclear power plant.

c. Training: Complete the Multi-Discipline Supervisor training program.

For the purpose of 10 CFR 55.4, a licensed senior reactor operator and a licensed reactor operator are those individuals who, in addition to meeting the requirements of 6.3.1, perform the functions described in 10 CFR 50.54(m).

6.4 TRAINING

6.4.1 Each member of the unit staff shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, Revision 3.

6.5 DELETED

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIST OF TABLES (Continued)

REACTIVITY CONTROL SYSTEMS

BORIC ACID MAKEUP PUMPS – OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 At least the boric acid makeup pump(s) in the boron injection flow path(s) required OPERABLE pursuant to Specification 3.1.2.2 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if the flow path through the boric acid pump(s) in Specification 3.1.2.2 is OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With no boric acid makeup pump required for the boron injection flow path(s) pursuant to Specification 3.1.2.2 operable, restore the boric acid makeup pump to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to its COLR limit at 200°F; restore the above required boric acid makeup pump(s) to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.6 The above required boric acid makeup pump(s) shall be demonstrated OPERABLE by verifying that the pump(s) develop the specified discharge pressure when tested pursuant to the Inservice Testing Program.

INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

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/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip 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 for the MODES and at the frequencies shown in Table 4.3-3.

4.3.3.2 At least once per 18 months, each Control Room Isolation radiation monitoring instrumentation channel shall be demonstrated OPERABLE by verifying that the response time of the channel is within limits.

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a.* With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown in Table 3.3-10, either restore the inoperable channel to OPERABLE status within 7 days, or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

b.* With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-10, either restore the inoperable channels to OPERABLE status within 48 hours or be in HOT STANDBY in 6 hours and HOT SHUTDOWN in 12 hours.

c.** With the number of OPERABLE Channels one less than the Total Number of Channels shown in Table 3.3-10, either restore the inoperable channel to OPERABLE status within 7 days if repairs are feasible without shutting down or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

d.** With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirements of Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

1. Initiate an alternate method of monitoring the reactor vessel inventory; and

2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status, and

3. Restore the Channel to OPERABLE status at the next scheduled refueling.

e. The provisions of Specification 3.0.4 are not applicable.

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* Action statements do not apply to Reactor Vessel Level Monitoring System, Containment Sump Water Level (narrow range) and Containment Sump Water Level (wide range) instruments.

** Action statements apply only to Reactor Vessel Level Monitoring System, Containment Sump Water Level (narrow range) and Containment Sump Water Level (wide range) instruments.

DELETED

6.0 ADMINISTRATIVE CONTROLS

6.4 TRAINING

6.4.1 Each member of the unit staff shall meet or exceed the minimum qualifications of Regulatory Guide 1.8, Revision 3.

6.5 *DELETED*

ADMINISTRATIVE CONTROLS (continued)

l. Steam Generator (SG) Program (continued)

1. (continued)

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outages nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary-to-secondary leakage

ADMINISTRATIVE CONTROLS (continued)

PAGES 6-15g AND 6-15h HAVE BEEN DELETED.
THE NEXT PAGE IS 6-15i.

m. Control Room Envelope Habitability Program

A Control Room Envelope (CRE) Habitability Program shall be established and implemented to ensure that CRE habitability is maintained such that, with an OPERABLE Control Room Emergency Air Cleanup System (CREACS), CRE occupants can control the reactor safely under normal conditions and maintain it in a safe condition following a radiological event, hazardous chemical release, or a smoke challenge. The program shall ensure that adequate radiation protection is provided to permit access and occupancy of the CRE under design basis accident (DBA) conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) for the duration of the accident.

The program shall include the following elements:

a. The definition of the CRE and the CRE boundary.

b. Requirements for maintaining the CRE boundary in its design condition including configuration control and preventive maintenance.

c. Requirements for (i) determining the unfiltered air inleakage past the CRE boundary into the CRE in accordance with the testing methods and at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," Revision 0, May 2003, and (ii) assessing CRE habitability at the Frequencies specified in Sections C.1 and C.2 of Regulatory Guide 1.197, Revision 0.

d. Measurement, at designated locations, of the CRE pressure relative to all external areas adjacent to the CRE boundary during the pressurization mode of operation by one train of the CREACS, operating at the flow rate required by the VFTP, at a Frequency of 36 months on a STAGGERED TEST BASIS. The results shall be trended and used as part of the 36 month assessment of the CRE boundary.

e. The quantitative limits on unfiltered air inleakage into the CRE. These limits shall be stated in a manner to allow direct comparison to the unfiltered air inleakage measured by the testing described in paragraph c. The unfiltered air inleakage limit for radiological challenges is the inleakage flow rate assumed in the licensing basis analyses of DBA consequences. Unfiltered air inleakage limits for hazardous chemicals must ensure that exposure of CRE occupants to these hazards will be within the assumptions in the licensing basis.

f. The provisions of SR 4.0.2 are applicable to the Frequencies for assessing CRE habitability, determining CRE unfiltered inleakage, and measuring CRE pressure and assessing the CRE boundary as required by paragraphs c and d, respectively.

ADMINISTRATIVE CONTROLS (continued)

n. 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:

(i) 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 or a water and sediment content within limits;

(ii) Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks; and

(iii) Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days.

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

o. Reactor Coolant Pump Flywheel Inspection Program

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

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC within the time period specified for each report.

6.10 DELETED