L. M. Stinson (Mike)
Vice President

Southern Nuclear
Operating Company, Inc.
40 Inverness Center Parkway
Post Office Box 1295
Birmingham, Alabama 35201

Tel 205.992.5181
Fax 205.992.0341

SOUTHERN COMPANY
Energy to Serve Your World℠

June 1, 2005

Docket Nos.: 50-348
50-364

NL-05-0328

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555-0001

Joseph M. Farley Nuclear Plant
Request for Technical Specification Amendment
Containment Tendon Surveillance Program

Ladies and Gentlemen:

In accordance with the provisions of 10 CFR 50.90, Southern Nuclear Operating Company (SNC) is submitting a request for an amendment to the Technical Specifications (TS) for Farley Nuclear Plant (FNP), Units 1 and 2.

The proposed amendment would revise the FNP TS section 5.5, "Programs and Manuals," section 5.6, "Reporting Requirements," and TS Bases for LCO 3.6.1, "Containment," relative to references of the FNP Containment Tendon Surveillance Program in order to reflect the latest requirements for tendon surveillance. By letter dated March 17, 2004, the NRC issued a similar amendment to Facility Operating License No. NPF-42 for the Wolf Creek Generating Station. The proposed changes are a result of the NRC issuing a final rule amending 10 CFR 50.55a, "Codes and Standards," which incorporated by reference the 1992 Edition with the 1992 Addenda of Subsection IWE, "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Plants," and Subsection IWL, "Requirements for Class CC Concrete Components of Light-Water Cooled Plants," of Section XI, Division 1, of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code).

Enclosure 1 provides the basis for the proposed TS change. Enclosure 2 provides the marked-up TS and TS Bases pages. Enclosure 3 provides the clean-typed TS and TS Bases pages.

SNC requests approval of the proposed license amendment by June 1, 2006, with the amendment being implemented within 90 days of issuance of the amendment.

A001

A copy of the proposed changes has been sent to Dr. D. E. Williamson, the Alabama State Designee, in accordance with 10 CFR 50.91(b)(1).

Mr. L. M. Stinson states he is a Vice President of Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company, and to the best of his knowledge and belief, the facts set forth in this letter are true.

This letter contains no NRC commitments. If you have any questions, please advise.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

L. M. Stinson

Sworn to and subscribed before me this 1 *day of* June *, 2005.*

Notary Public

My commission expires: 4-28-07

LMS/LPH/sdl

Enclosures:
1. Basis for the Proposed Change
2. Marked-Up Technical Specifications and Bases Pages
3. Clean-Typed Technical Specifications and Bases Pages

cc: Southern Nuclear Operating Company
Mr. J. T. Gasser, Executive Vice President
Mr. J. R. Johnson, General Manager – Plant Farley
RTYPE: CFA04.054; LC# 14223

U. S. Nuclear Regulatory Commission
Dr. W. D. Travers, Regional Administrator
Mr. R. E. Martin, NRR Project Manager – Farley
Mr. C. A. Patterson, Senior Resident Inspector – Farley

Alabama Department of Public Health
Dr. D. E. Williamson

Joseph M. Farley Nuclear Plant
Request for Technical Specifications Amendment
Containment Tendon Surveillance Program

Enclosure 1

Basis for the Proposed Change

Joseph M. Farley Nuclear Plant
Request for Technical Specifications Amendment
Containment Tendon Surveillance Program

Enclosure 1

Basis for the Proposed Change

1.0 Description

The proposed change revises the Farley Nuclear Plant (FNP) Units 1 and 2 Technical Specifications (TS) 5.5, "Programs and Manuals," TS 5.6, "Reporting Requirements," and TS Bases for LCO 3.6.1, "Containment," to reflect the latest requirements for tendon surveillance. The proposed changes are a result of the NRC issuing a final rule amending 10 CFR 50.55a that affected the surveillance methods for the containment tendons and the conduct of containment visual inspections, and the methods of reporting the results of the required inspections to the NRC. These revised requirements were to be fully implemented by September 9, 2001.

2.0 Proposed Change

The following change (in italics) is proposed for the last sentence of the first paragraph in TS section 5.5.6, "Pre-Stressed Concrete Containment Tendon Surveillance Program:"

> "The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with *Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, except where an alternative, exemption or relief has been authorized by the NRC. The first performance of the IWL requirements for containment sample tendon force measurements and tendon wire and strand sample examinations will be performed by the end of 2006.*"

In addition, since the tendon inspection frequencies will be in accordance with ASME Section XI, Subsection IWL, the provisions of SR 3.0.2 are no longer applicable; therefore, deletion of the provisions of SR 3.0.2 from Technical Specification 5.5.6 is also proposed. 10 CFR 50.55a requires the implementation of ASME Section XI, Subsection IWL and specifies the requirements for extending inspection frequencies.

Also, since reporting requirements for the Tendon Surveillance Report will be in accordance with the reporting requirements in 10 CFR 50.55a, deletion of the provisions of Technical Specification 5.6.9 is also proposed. Repeating the Federal Regulations within the TS is not necessary to ensure safe operation of the plant. Therefore, the deletion of this TS requirement is acceptable.

The following change (in italics) is proposed for the last sentence in TS Bases for LCO 3.6.1, "Containment," SR 3.6.1.2:

> "Testing and Frequency are consistent with the *requirements of Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable*

addenda as required by 10 CFR 50.55a, except where an alternative, exemption or relief has been authorized by the NRC (Ref. 4)."

The following change (in italics) is proposed for TS Bases for LCO 3.6.1, "Containment," in the "References" section:

Revise reference #4 to: *Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a.*

The current TS and TS Bases refer to Regulatory Guide 1.35, Revision 2.

3.0 Background

Prior to September 26, 2000, post tensioning exams conducted at Farley Nuclear Plant (FNP) were performed in accordance with Regulatory Guide (RG) 1.35, Rev. 2 in accordance with the plant Technical Specifications.

Since both FNP units had similar containments, RG 1.35, Rev. 2 allowed containment sample tendon force measurements and tendon wire and strand sample examinations to be performed every 5 years (based on the Structural Integrity Test (SIT) date) on Unit 1 only. Visual examinations and exams for water at the corrosive protective medium were required on the tendon anchorage area for each unit every 5 years.

Containment sample tendon force measurements and tendon wire and strand sample (full) examinations were performed on Unit 1 in 1978 (1 year exams), 1980 (3 year exams), 1982 (5 year exams), 1987 (10 year exams) and in 1992 (15 year exams). The next examination was due in 1997 (20 year exams). Visual examinations and exams for water at the corrosive protective medium were performed every five years on each unit as required.

A rule change to 10CFR 50.55a dated August 8, 1996, required licensees to perform expedited Containment examinations in accordance with the 1992 edition and addenda of Section XI, Subsection IWL. The expedited examination schedule required the first exams be completed by September 9, 2001. However, for containment sample tendon force measurements and tendon wire and strand sample examinations, the rulemaking gave licensees the option to perform the expedited examinations (i.e. exams performed between September 9, 1996 and September 9, 2001) in accordance with a program that was approved by the NRC prior to September 9, 1996 (e.g., the Technical Specifications).

FNP elected to continue with the Technical Specification/RG 1.35 examinations and performed the 20 year Unit 1 containment sample tendon force measurements and tendon wire and strand sample examinations in 1997. All acceptance criteria were met. The next required examination for Unit 1 was in 2002 (25 year exams).

Joseph M. Farley Nuclear Plant
Request for Technical Specifications Amendment
Containment Tendon Surveillance Program

Enclosure 1

On September 26, 2000 the Containment Inspection Plan, which is the scheduling device for performing FNP Containment exams, was revised to incorporate the requirements of IWL. This revision deleted the TS/RG 1.35 schedule; however, the TS (SR 3.6.1.2 and TS 5.5.6) were not changed, thus they remained in effect.

IWL-2420(a) required containment sample tendon force measurements and tendon wire and strand sample examinations on each unit every 5 years (based on SIT). Visual examinations and exams for water were required on the tendon anchorage area for each unit every 5 years per IWL-2420. For the 10 year and subsequent examinations, IWL-2420(c) allows examinations to be scheduled ± 1 year from the specified schedule date. Visual examinations were performed in 2000 on FNP Unit 2 and in 2002 on FNP Unit 1. For each of these surveillances, all acceptance criteria were met.

However, IWL-2421 provided an option that allowed extending containment sample tendon force measurements and tendon wire and strand sample examinations on both units to a 10 year frequency if both are similar and if both post tensioning operations were completed not more than 2 years apart. Visual examinations and exams for water at the corrosive protective medium are required on the tendon anchorage area for each unit every 5 years.

At FNP, the post tension operations were performed 26 months apart. Therefore, the IWL-2421(a) requirements were not met. Even though this exceeds the requirements of IWL-2421(a) for applying IWL-2421(b) by 2 months, exceeding the post tensioning operations required time by only 2 months does not present any safety or technical concerns because a 2 month difference is insignificant. Both units share the same design, were under construction during the same time period in a sequential manner, and are exposed to virtually identical environmental conditions. However, FNP inappropriately interpreted the allowance of IWL-2421(a) and elected to use the IWL-2421(b) option and deferred the Unit 1 25 year containment sample tendon force measurements and tendon wire and strand sample examinations due in 2002 until 2007 (30 year exams) instead of performing them in 2002 as required by IWL-2420. In addition, since the TS still referenced RG 1.35, Rev. 2 as the basis for the Pre-Stressed Concrete Containment Tendon Surveillance Program (TS 5.5.6), and RG 1.35, Rev. 2 required an examination in 2002 as well, TS SR 3.6.1.2 was missed.

TS 5.5.6 states that SR 3.0.3 is applicable to the Tendon Surveillance Program inspection frequencies. SR 3.0.3 states that if it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed. In addition, the Bases state that all missed Surveillances will be placed in the licensee's Corrective Action Program. FNP has completed a qualitative risk evaluation with regard to the risk of delaying the missed examination and has documented the evaluation in its Corrective Action

Program. The risk evaluation concluded that it is acceptable from a risk perspective to delay the performance of the containment sample tendon force measurements and tendon wire and strand sample examinations for Unit 1 until the planned performance in 2006.

SNC has submitted in letter NL-05-0285 a request for alternative, RR-57, requesting permission from the NRC to use the 10 year scheduling option (even though the post tensioning operations were performed slightly greater than 2 years apart). This option will allow FNP to use the IWL-2421(b) inspection requirements for two site containments whose post tensioning operations were completed not more than 2 years apart. In addition, letter NL-05-0285 includes a request for alternative, RR-58, regarding future IWL containment examination dates based on a common administrative date of July 2006 for both FNP units.

The scope of this proposed change updates the TS and TS Bases to reflect the latest requirements of the Containment Tendon Surveillance Program that FNP has incorporated into its ISI program.

4.0 Regulatory Analysis

No Significant Hazards Consideration

The proposed change revises the Farley Nuclear Plant (FNP) Units 1 and 2 Technical Specifications (TS) 5.5, "Programs and Manuals," TS 5.6, "Reporting Requirements," and TS Bases for LCO 3.6.1, "Containment," to reflect the latest requirements for tendon surveillance. Since the tendon inspection frequencies will be in accordance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code) Section XI, Subsection IWL, the provisions of SR 3.0.2 are no longer applicable and are deleted from Technical Specification 5.5.6. 10 CFR 50.55a requires the implementation of ASME Section XI, Subsection IWL and specifies the requirements for extending inspection frequencies.

In TS 5.5.6, reference to the first performance of the IWL requirements for containment sample tendon force measurements and tendon wire and strand sample examinations to be performed by the end of 2006 is proposed. In addition, since reporting requirements are contained in 10 CFR 50.55a, deletion of the provisions of Technical Specification 5.6.9 is proposed. Repeating the Federal Regulation within the TS is not necessary to ensure safe operation of the plant. Therefore, the deletion of this TS requirement is acceptable. Also, several editorial changes in the TS are proposed.

Southern Nuclear Operating Company (SNC) has evaluated whether or not a significant hazards consideration is involved with the proposed change by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

Joseph M. Farley Nuclear Plant
Request for Technical Specifications Amendment
Containment Tendon Surveillance Program

Enclosure 1

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

   The proposed change replaces the previous TS requirement to implement a Containment Tendon Surveillance Program based on Regulatory Guide 1.35, Rev. 2, with a Containment Inspection Program that complies with the current requirements of 10 CFR 50.55a. This regulation requires licensees to implement a Containment Inspection Program in compliance with the 1992 Edition with the 1992 Addenda of Subsection IWE, "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Plants," and with Subsection IWL, "Requirements for Class CC Concrete Components of Light-Water Cooled Plants," of Section XI, Division 1, of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) with additional modifications and limitations as stated in 10 CFR 50.55a(b)(2)(ix). SNC has implemented a Containment Inspection Program that complies with the regulatory requirements. This proposed TS amendment is requested to update the TS to the latest 10 CFR 50.55a regulatory requirements.

   In addition, reporting requirements that are redundant to existing regulations are deleted, minor editorial changes are made, and the applicability of SR 3.0.2 to the tendon surveillance program is deleted since surveillance frequencies and associated extensions are specified in ASME Section XI, Subsection IWL.

   By complying with the regulatory requirements described in 10 CFR 50.55a, the probability of a loss of containment structural integrity is maintained as low as reasonably achievable. Maintaining containment structural integrity as described in the revised Containment Inspection Program does not impact the operation of the reactor coolant system (RCS), containment spray (CS) system, or emergency core cooling system (ECCS). The Containment Inspection Program ensures that the containment will function as designed to provide an acceptable barrier to release of radioactive materials to the environment. The proposed change does not alter or prevent the ability of structures, systems, and components (SSCs) from performing their intended function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed change does not impact any accident initiators or analyzed events, nor does it impact the types or amounts of radioactive effluent that may be released offsite. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Maintaining containment structural integrity does not impact the operation of the RCS, CS system, or ECCS. The proposed change does not involve a modification to the physical configuration of the plant or a change in the methods governing normal plant operation. The proposed change does not introduce a new accident initiator, accident precursor, or malfunction mechanism. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed license amendment does not involve a significant reduction in a margin of safety.

   By complying with the regulatory requirements described in 10 CFR 50.55a, the probability of a loss of containment structural integrity is maintained as low as reasonably achievable. The Containment Inspection Program ensures that the containment will function as designed to provide an acceptable barrier to release of radioactive materials to the environment. The proposed change does not adversely affect plant operation or existing safety analyses. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

5.0 Environmental Considerations

SNC has reviewed the proposed change pursuant to 10 CFR 50.92 and determined that it does not involve a significant hazards consideration. In addition, there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite, and there is no significant increase in individual or cumulative occupational radiation exposure. Consequently, the proposed TS change has no significant effect on the human environment and satisfies the criteria of 10 CFR 51.22 for categorical exclusion from the requirements for an environmental assessment.

Joseph M. Farley Nuclear Plant
Request for Technical Specifications Amendment
Containment Tendon Surveillance Program

Enclosure 2

Marked-Up Technical Specifications and Bases Pages

TS Page 5.5-4
TS Page 5.6-5
TS Page 5.6-6
TS Bases Page B 3.6.1-5

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

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

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

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the FSAR, Table 5.2-2a, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with ~~Regulatory Guide 1.35, Revision 2, 1976~~ INSERT 1

The provisions of ~~SR 3.0.2 and~~ SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel at least once per 10 years by conducting either:

a. An in-place ultrasonic examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius: or

b. A surface examination (magnetic particle and/or liquid penetrant) of exposed surfaces of the disassembled flywheel.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Reactor Coolant Pump Flywheel Inspection Program.

INSERT 1

Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, except where an alternative, exemption or relief has been authorized by the NRC. The first performance of the IWL requirements for containment sample tendon force measurements and tendon wire and strand sample examinations will be performed by the end of 2006.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. The reactor coolant system pressure and temperature limits, including heatup and cooldown rates, shall be established and documented in the PTLR for LCO 3.4.3.

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the NRC letters dated March 31, 1998 and April 3, 1998.

c. The PTLR shall be provided to the NRC upon issuance for each reactor fluence period and for any revision or supplement thereto.

5.6.7 EDG Failure Report

If an individual emergency diesel generator (EDG) experiences four or more valid failures in the last 25 demands, these failures shall be reported within 30 days. Reports on EDG failures shall include a description of the failures, underlying causes, and corrective actions taken per the Emergency Diesel Generator Reliability Monitoring Program.

5.6.8 PAM Report

When a report is required by Condition B or ~~G~~ F of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.9 ~~Tendon Surveillance Report~~ Deleted.

~~Any abnormal degradation of the containment structure detected during the tests required by the Pre-stressed Concrete Containment Tendon Surveillance~~

(continued)

5.6 Reporting Requirements

~~5.6.9~~ ~~Tendon Surveillance Report (continued)~~

~~Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.~~

5.6.10 Steam Generator (SG) Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.9, Steam Generator (SG) Program. 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 during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged to date, and

g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

5.6.11 Alternate AC (AAC) Source Out of Service Report

The NRC shall be notified if the AAC source is out of service for greater than 10 days.

BASES

SURVEILLANCE REQUIREMENTS ~~ACTIONS~~ (continued)

SR 3.6.1.2

For ungrouted, post tensioned tendons, this SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are consistent with the ~~recommendations~~ requirements of ~~Regulatory Guide 1.35~~ Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, except where an alternative, exemption or relief has been authorized by the NRC (Ref. 4).

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. FSAR, Chapter 15.
3. FSAR, Section 6.2.
4. ~~Regulatory Guide 1.35, Revision 2~~ Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a.

Joseph M. Farley Nuclear Plant
Request for Technical Specifications Amendment
Containment Tendon Surveillance Program

Enclosure 3

Clean-Typed Technical Specifications and Bases Pages

TS Page 5.5-4
TS Page 5.5-5
TS Page 5.5-6
TS Page 5.6-5
TS Page 5.6-6
TS Bases Page B 3.6.1-5

5.5.4 Radioactive Effluent Controls Program (continued)

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

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

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the FSAR, Table 5.2-2a, cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, except where an alternative, exemption or relief has been authorized by the NRC. The first performance of the IWL requirements for containment sample tendon force measurements and tendon wire and strand sample examinations will be performed by the end of 2006.

The provisions of SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel at least once per 10 years by conducting either:

a. An in-place ultrasonic examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius: or

b. A surface examination (magnetic particle and/or liquid penetrant) of exposed surfaces of the disassembled flywheel.

(continued)

5.5 Programs and Manuals

5.5.7 Reactor Coolant Pump Flywheel Inspection Program (continued)

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Reactor Coolant Pump Flywheel Inspection Program.

5.5.8 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. 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 SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

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

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

5.5.9 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring

(continued)

5.5.9 Steam Generator (SG) Program (continued)

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 an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

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 inservice 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 (3ΔP) 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. Accident induced leakage is not to exceed 1 gpm total for all three SGs.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. The reactor coolant system pressure and temperature limits, including heatup and cooldown rates, shall be established and documented in the PTLR for LCO 3.4.3.

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the NRC letters dated March 31, 1998 and April 3, 1998.

c. The PTLR shall be provided to the NRC upon issuance for each reactor fluence period and for any revision or supplement thereto.

5.6.7 EDG Failure Report

If an individual emergency diesel generator (EDG) experiences four or more valid failures in the last 25 demands, these failures shall be reported within 30 days. Reports on EDG failures shall include a description of the failures, underlying causes, and corrective actions taken per the Emergency Diesel Generator Reliability Monitoring Program.

5.6.8 PAM Report

When a report is required by Condition B or F of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.9 Deleted

5.6 Reporting Requirements

5.6.10 Steam Generator (SG) Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.9, Steam Generator (SG) Program. 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 during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged to date, and

g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

5.6.11 Alternate AC (AAC) Source Out of Service Report

The NRC shall be notified if the AAC source is out of service for greater than 10 days.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.2

For ungrouted, post tensioned tendons, this SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are consistent with the requirements of Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, except where an alternative, exemption or relief has been authorized by the NRC (Ref. 4).

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. FSAR, Chapter 15.
3. FSAR, Section 6.2.
4. Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a.