Entergy

Entergy Operations, Inc.
17265 River Road
Killona, LA 70066
Tel 504 739 6650

W3F1-2006-0029

June 2, 2006

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

SUBJECT: Revision to Amendment Request NPF-38-262
Steam Generator Tube Inservice Inspection Program
Waterford Steam Electric Station, Unit 3
Docket No. 50-382
License No. NPF-38

REFERENCES:
1. Entergy letter dated July 21, 2005, *License Amendment Request NPF-38-262 Proposed Technical Specification Change to Waterford-3 Steam Generator Tube Inservice Inspection Program Using Consolidated Line Item Improvement Process* (W3F1-2005-0040)
2. Entergy letter dated February 15, 2006, *Supplement to Amendment Request NPF-38-262 Steam Generator Tube Inservice Inspection Program* (W3F1-2006-0007)
3. Entergy letter dated May 3, 2006, *Supplement 2 to Amendment Request NPF-38-262 Steam Generator Tube Inservice Inspection Program* (W3F1-2006-0016)

Dear Sir or Madam:

By letter dated July 21, 2005 (Reference 1), Entergy Operations, Inc. (Entergy) proposed a change to the Waterford Steam Electric Station, Unit 3 (Waterford-3) Technical Specifications (TSs) to replace the existing steam generator tube surveillance program with that being proposed by the Technical Specification Task Force in TSTF-449, Revision 4. TSTF-449, Revision 4 is formatted to the Improved Technical Specification (ITS) plants while the Waterford-3 TSs is based on the CE standard TSs. Therefore, the information contained in TSTF-449, Revision 4 was modified to correspond with the Waterford-3 TS format.

On January 3, 2006, Entergy received an NRC Staff Request for Additional Information (RAI) on the proposed amendment request. The RAI response was provided on February 15, 2006 in Reference 2. On April 17, 2006, Entergy received a second NRC Staff RAI dated March 31, 2006 on the proposed amendment request. The RAI response was provided on May 3, 2006 in Reference 3.

A047

On May 18, 2006, Entergy and members of the NRC staff held a call to discuss correction of certain changes to the Waterford-3 TSs identified in the May 3, 2006 submittal (Reference 3). As a result of the discussion, Entergy agreed to provide a follow-up letter with the appropriate corrections.

Attachment 1 lists the changes that are being made to the TS and associated Bases since the May 3, 2006 submittal. Attachment 2 provides the revised clean TS pages. The TS Bases pages will be incorporated into the Waterford-3 TS Bases through the TS Bases Control Program upon implementation of the license amendment.

The conclusions of the original no significant hazards consideration included in Reference 1 are not affected by any information contained in this supplemental letter. There are no new commitments contained in this letter.

If you have any questions or require additional information, please contact Steve Bennett or Ron Williams at (479) 858-4626 and (504) 739-6255, respectively.

I declare under penalty of perjury that the foregoing is true and correct. Executed on June 2, 2006.

Sincerely,

R.J. Murillo
Acting Nuclear Safety Assurance
Director

RJM/RLW

Attachments:
1. List of Proposed Technical Specification and Bases Changes
2. Technical Specification Clean Pages

cc: Dr. Bruce S. Mallett
Regional Administrator
U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-8064

NRC Senior Resident Inspector
Waterford Steam Electric Station Unit 3
P.O. Box 822
Killona, LA 70066-0751

U.S. Nuclear Regulatory Commission
Attn: Mr. Mel B. Fields MS O-7E1
Washington, DC 20555-0001

Wise, Carter, Child & Caraway
ATTN: J. Smith
P.O. Box 651
Jackson, MS 39205

Winston & Strawn
ATTN: N.S. Reynolds
1700 K Street, NW
Washington, DC 20006-3817

Morgan, Lewis & Bockius LLP
ATTN: T.C. Poindexter
1111 Pennsylvania Avenue, NW
Washington, DC 20004

Louisiana Department of Environmental Quality
Office of Environmental Compliance
Surveillance Division
P. O. Box 4312
Baton Rouge, LA 70821-4312

American Nuclear Insurers
Attn: Library
Town Center Suite 300S
29th S. Main Street
West Hartford, CT 06107-2445

Attachment 1
To
W3F1-2006-0029

List of Proposed Technical Specification and Bases Changes

List of Proposed Technical Specification and Bases Changes

The following are proposed changes to the Technical Specification (TS) pages in support of the Waterford-3 license amendment for the Steam Generator Tube Inservice Inspection Program. The proposed changes reflect correction of the NRC and Entergy identified editorial errors since the May 3, 2006 letter (Reference 3). The TS clean pages, that contain all the proposed changes, are included in Attachment 2 of this letter.

- TS definition 1.14 c. – added parentheses around "primary to secondary leakage." It now reads "Reactor Coolant System leakage through a steam generator to the secondary system (primary to secondary leakage)."
- TS 3.4.4 - Added ", and" after the first LCO. It now reads "SG tube integrity shall be maintained, and ..." (consistent with TSTF-449).
- TS 3.4.4 - Added "NOTE:" before the sentence that begins with Separate.... (consistent with TSTF-449). In the same sentence Action was capitalized to ACTION since it is a defined term. (Similar changes were made in the Bases.)
- TS 3.4.4 - In ACTION a, "are" was removed from the statement where it now reads "...repair criteria and not plugged..." (consistent with TSTF-449).
- TS 3.4.4 - In ACTION b, the term Action was capitalized to ACTION in two places since it is a defined term. Deleted the period "." after "Action a". It now reads "...Outage Time of ACTION a above." In addition, "the" was removed prior to SG tube integrity. It now reads "...cannot be met or SG tube integrity cannot be maintained..." (consistent with TSTF-449). Also corrected the editorial error where "with" was used instead of "within." It now reads "...COLD SHUTDOWN within the following 30 hours."
- TS page 3/4 4-11 was added to replace the deleted sections of TS 3.4.4 and to indicate that TS pages 3/4 4-12 through 3/4 4-16 have been deleted and the next page being 3/4 4-17.
- TS 3.4.5.2 c. - Spelled out steam generator after the phrase "through any one" and placed "SG" in parenthesis. It now reads "... through any one steam generator (SG)."
- TS 3.4.5.2 – In ACTION a, the word "any" was removed from any primary to secondary leakge. The statement now reads "With any PRESSURE BOUNDARY LEAKAGE, or primary to secondary leakage not within limit...."
- TS 6.5.9 - Item b.3, "operational leakage" was capitalized in the reference to LCO 3.4.5.2, *Reactor Coolant System Operational Leakage.*
- TS 6.5.9 - Item d.1, "[percent]" was removed from the statements where it now reads "Inspect 100% of the .... ."

The following are the proposed changes to the TS Bases pages in support of the Waterford-3 license amendment for the Steam Generator Tube Inservice Inspection Program. The Bases pages will be incorporated into the Waterford-3 TS Bases through the TS Bases Control Program upon implementation of the license amendment.

- TS Bases 3/4.4.5 - In the third paragraph of the "Safety Analysis" section, References 2 and 3 were added (consistent with TSTF-449).
- TS Bases 3/4.4.5 – In the second paragraph of the first bullet under "Limiting Condition for Operation" section, References 4 and 5 were added (consistent with TSTF-449).
- TS Bases 3/4.4.5 - In the last paragraph under ACTIONS, the word "time" was pluralized to now read "The allowed outage time<u>s</u> are reasonable" (consistent with TSTF-449).

On multiple pages of both the TS and TS Bases, the use of punctuation and quotation marks (") were modified where appropriate.

Attachment 2
To
W3F1-2006-0029

Technical Specification Clean Pages (10)

IDENTIFIED LEAKAGE (Continued)

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the secondary system (primary to secondary leakage).

MEMBER(S) OF THE PUBLIC

1.15 MEMBER(S) OF THE PUBLIC means any individual except when that individual is receiving an occupational dose.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.16 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specification 6.9.1.7 and 6.9.1.8.

OPERABLE - OPERABILITY

1.17 A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.18 An OPERATIONAL MODE (i.e. MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.2.

PHYSICS TEST

1.19 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PLANAR RADIAL PEAKING FACTOR - Fxy

1.20 The PLANAR RADIAL PEAKING FACTOR is the ratio of the peak to plane average power density of the individual fuel rods in a given horizontal plane, excluding the effects of azimuthal tilt.

PRESSURE BOUNDARY LEAKAGE

1.21 PRESSURE BOUNDARY LEAKAGE shall be leakage (except primary to secondary leakage) through a non isolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

PROCESS CONTROL PROGRAM (PCP)

1.22 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, state regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

1.23 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity,concentra- or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

REACTOR COOLANT SYSTEM

3/4.4.4 STEAM GENERATOR (SG) TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.4

a. SG tube integrity shall be maintained, and

b. All SG tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program.

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

ACTION:

NOTE: Separate ACTION entry is allowed for each SG tube.

a. With one or more SG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program.

   1. Within 7 days verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection, and

   2. Plug the affected tube(s) in accordance with the Steam Generator Program prior to entering HOT SHUTDOWN following the next refueling outage or SG tube inspection.

b. If the required ACTION and Allowed Outage Time of ACTION a above cannot be met or SG tube integrity cannot be maintained, be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.4.1 Verify SG tube integrity in accordance with the Steam Generator Program.

4.4.4.2 Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program prior to entering HOT SHUTDOWN following a SG tube inspection.

Pages 3/4 4-12 through 3/4 4-16 have been deleted.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.5.2 Reactor Coolant System operational leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 75 gallons per day primary to secondary leakage through any one steam generator (SG),

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 1 gpm leakage at a Reactor Coolant System pressure of 2250 ± 20 psia from any Reactor Coolant System pressure isolation valve specified in Table 3.4-1.

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

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, or primary to secondary leakage not within limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any Reactor Coolant System operational leakage greater than any one of the limits, excluding PRESSURE BOUNDARY LEAKAGE, primary to secondary leakage, and leakage from Reactor Coolant System pressure isolation valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one closed manual or deactivated automatic valve, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

NOTE: Not required to be performed until 12 hours after establishment of steady state operation.

4.4.5.2.1 Reactor Coolant System leakages, except for primary to secondary leakage, shall be demonstrated to be within each of the above limits by performance of a Reactor Coolant System water inventory balance at least once per 72 hours.

4.4.5.2.2 Primary to secondary leakage shall be verified to be $\leq$ 75 gallons per day through any one SG at least once per 72 hours.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.2.3 Each Reactor Coolant System pressure isolation valve specified in Table 3.4-1, Section A and Section B, shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 7 days or more and if leakage testing has not been performed in the previous 9 months,

c. Prior to returning the valve to service following maintenance, repair, or replacement work on the valve,

d. Following valve actuation for valves in Section B due to automatic or manual action or flow through the valve:

   1. Within 24 hours by verifying valve closure, and

   2. Within 31 days by verifying leakage rate.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

4.4.5.2.4 Each Reactor Coolant System pressure isolation valve power-operated valve specified in Table 3.4-1, Section C, shall be demonstrated OPERABLE by verifying leakage to be within its limit:

a. At least once per 18 months, and

b. Prior to returning the valve to service following maintenance, repair, or replacement work on the valve.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

6.5.8 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 Specification 4.0.2 are applicable to the above required frequencies for performing inservice testing activities.

c. The provisions of Specification 4.0.3 are applicable to inservice testing activities, and

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

6.5.9 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 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.

STEAM GENERATOR (SG) PROGRAM (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 steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down 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. Primary to secondary leakage is not to exceed 540 gpd through any one SG.

3. The operational leakage performance criterion is specified in LCO 3.4.5.2, "Reactor Coolant System Operational Leakage."

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.

STEAM GENERATOR (SG) PROGRAM (Continued)

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 employs and at what locations.

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

2. Inspection 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. 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.

ANNUAL REPORTS (Continued)

(1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded;

(2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations;

(3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded;

(4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above steady-state level; and

(5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

6.9.1.5 STEAM GENERATOR TUBE INSPECTION REPORT

A report shall be submitted within 180 days after the initial entry into HOT SHUTDOWN following completion of an inspection performed in accordance with the Specification 6.5.9, 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,

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

h. The effective plugging percentage for all plugging in each SG.