Entergy

Entergy Operations, Inc.
1448 S.R. 333
Russellville, AR 72802
Tel 479-858-4601

Thomas A. Marlow
Director,
Nuclear Safety Assurance

2CAN060602

June 19, 2006

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

SUBJECT: License Amendment Request
Revised Technical Specification Pages for ANO-2 Steam Generator Tube Inservice Inspection Program
Arkansas Nuclear One, Unit 2
Docket No. 50-368
License No. NPF-6

REFERENCE: 1 Entergy letter dated September 19, 2005, *Proposed Technical Specification Change to ANO-2 Steam Generator Tube Inservice Inspection Program Using Consolidated Line Item Improvement Process* (2CAN090501)

2 Entergy letter dated May 11, 2006, Response to NRC Request for Additional Information on ANO-2 Steam Generator Tube Inservice Inspection Program (2CAN050601)

Dear Sir or Madam:

In Reference 1, Entergy Operations, Inc. (Entergy) requested an Operating License (OL) amendment for Arkansas Nuclear One, Unit 2 (ANO-2) 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 ANO-2 technical specifications (TS) are based on the CE standard TSs. Therefore, the information contained in TSTF-449, Revision 4 was modified to correspond with the ANO-2 TS format.

On March 30, 2006, Entergy received Requests for Additional Information (RAI) from the NRC staff on the proposed amendment request. The RAI response was provided on May 11, 2006 in Reference 2. Since the May 11, 2006 submittal several corrections have been identified which require revision to the TS pages and associated Bases. Attachment 1 identifies the changes that are being made to the ANO-2 TSs and associated Bases. Attachment 2 provides the revised TS pages. In accordance with the ANO-2 TS Bases Control Program, the Bases pages will be incorporated into the ANO-2 TS Bases upon implementation of the license amendment.

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The proposed changes to the ANO-2 TSs do not impact the original No Significant Hazards Considerations contained in Reference 1. There are no new commitments associated with this letter.

If you have any questions or require additional information, please contact Steve Bennett at 479-858-4626.

I declare under penalty of perjury that the foregoing is true and correct. Executed on June 19, 2006.

Sincerely,

for

TAM/sab

Attachments:

1. Proposed Technical Specification and Bases Changes to ANO-2 OL Amendment on Steam Generator Tube Inservice Inspection Program
2. Proposed Technical Specification Changes

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
Arkansas Nuclear One
P. O. Box 310
London, AR 72847

U. S. Nuclear Regulatory Commission
Attn: Mr. Drew Holland MS O-7D1
Washington, DC 20555-0001

Arkansas Department of Health & Human Services
Division of Health
P.O. Box 1437
Slot H-30
Little Rock, AR 72203-1437

Attachment 1

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Proposed Technical Specification and Bases Changes to ANO-2 OL Amendment on Steam Generator Tube Inservice Inspection Program

Proposed Technical Specification and Bases Changes to ANO-2 OL Amendment on Steam Generator Tube Inservice Inspection Program

The following are proposed changes to the Technical Specification (TS) pages in support of the ANO-2 license amendment for the Steam Generator (SG) Tube Inspection Program. These changes are editorial and do not require additional justification. The revised TS pages are contained in Attachment 2 of this letter.

- TS 3.4.5 - Added ", and" after the first Limiting Condition for Operation (LCO). The LCO now reads "SG tube integrity shall be maintained, and" (consistent with TSTF-449)
- TS 3.4.5 - The term Action was capitalized to ACTION since it is a defined term. (Similar changes will be made in the TS Bases)
- The sentence "ACTIONS may be entered separately for each SG tube" has been identified as a note (consistent with TSTF-449).
- TS 3.4.5 - In ACTION a, "are" was removed from the statement where the ACTION now reads "...repair criteria and not plugged..." (consistent with TSTF-449)
- TS 3.4.5 - ACTION b, "the" was removed prior to SG tube integrity. The ACTION now reads "...cannot be met or SG tube integrity cannot be maintained..." (consistent with TSTF-449)
- TS 6.5.9 - Item b.3, "operational leakage" was capitalized in the reference to LCO 3.4.6.2, *Reactor Coolant System Operational Leakage*
- TS 6.5.9 - Item d.2, the bracket at the end of the sentence was removed
- TS 6.6.7 - In reporting requirement g, "and" was added at the end of the sentence. The reporting requirement now reads ...and in-situ testing, and... (consistent with TSTF-449)

The following are the proposed changes the TS Bases pages in support of the ANO-2 license amendment for the Steam Generator Tube Inspection Program. In accordance with the ANO-2 TS Bases Control Program, the Bases pages will be incorporated into the ANO-2 TS Bases upon implementation of the license amendment.

- TS Bases 3/4.4.5 - Under the Safety Analysis section the reference to the ANO-2 SAR was spelled out and "Section" was added. It now reads, "See Safety Analysis Report Section 15.1.18"
- TS Bases 3/4.4.5 - References 4 and 5 were added to the second paragraph of the first bullet under "Limiting Condition for Operation" (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 times are reasonable, ..." (consistent with TSTF-449)

Attachment 2

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Proposed Technical Specification Changes

CHANNEL FUNCTIONAL TEST

1.11 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels – The injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

b. Bistable channels – The injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

c. Digital computer channels – The exercising of the digital computer hardware using diagnostic programs and the injection of simulated process data into the channel to verify OPERABILITY.

CORE ALTERATION

1.12 CORE ALTERATION shall be the movement or manipulation of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATION shall not preclude completion of movement of a component to a safe conservative position.

SHUTDOWN MARGIN

1.13 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all control element assemblies are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn.

IDENTIFIED LEAKAGE

1.14 IDENTIFIED LEAKAGE shall be:

a. Leakage (except controlled leakage) into closed systems, such as pump seal or valve packing leaks that are captured, and conducted to a sump or collecting tank, or

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).

UNIDENTIFIED LEAKAGE

1.15 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or controlled leakage.

PRESSURE BOUNDARY LEAKAGE

1.16 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.

AZIMUTHAL POWER TILT – T_q

1.17 AZIMUTHAL POWER TILT shall be the power asymmetry between azimuthally symmetric fuel assemblies.

DOSE EQUIVALENT I-131

1.18 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.19 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MEV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

STAGGERED TEST BASIS

1.20 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

FREQUENCY NOTATION

1.21 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

STEAM GENERATOR (SG) TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.5 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: ACTIONS may be entered separately 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

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

4.4.5.2 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.

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System operational leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 150 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. Leakage as specified in Table 3.4.6-1 for those Reactor Coolant System Pressure Isolation Valves identified in Table 3.4.6.1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE or any 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 above limits, excluding PRESSURE BOUNDARY LEAKAGE and primary to secondary leakage, 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 two valves* in each high pressure line having a non-functional valve and be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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* These valves may include check valves for which the leakage rate has been verified, manual valves or automatic valves. Manual and automatic valves shall be tagged as closed to preclude inadvertent valve opening.

REACTOR COOLANT SYSTEM OPERATIONAL LEAKAGE

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System operational leakage, except for primary to secondary leakage, shall be demonstrated to be within each of the above limits by:

a. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady state operation except when operating in the shutdown cooling mode*.

b. Monitoring the reactor head flange leakoff temperature at least once per 24 hours.

4.4.6.2.2 Primary to secondary leakage shall be verified to be ≤ 150 gallons per day through any one SG at least once per 72 hours*.

4.4.6.2.3 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4.6-1 shall be demonstrated OPERABLE by individually verifying leakage to be within its limit:

a. Prior to entering MODE 2 after each refueling outage,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months, and

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

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

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.

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. Leakage is not to exceed 1 gpm through any one SG.

    3. The operational leakage performance criterion is specified in LCO 3.4.6.2, *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.

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 outage 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.

Next Page is 6-14

6.6.6 not used

6.6.7 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.

6.6.8. Specific Activity

The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (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 the 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 the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.