August 10, 2005

Mr. Jeffrey S. Forbes
Site Vice President
Arkansas Nuclear One
Entergy Operations, Inc.
1448 S. R. 333
Russellville, AR 72801

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: STEAM GENERATOR TUBE INSERVICE INSPECTION PROGRAM (TAC NO. MC4558)

Dear Mr. Forbes:

The Commission has issued the enclosed Amendment No. 224 to Renewed Facility Operating License No. DPR-51 for Arkansas Nuclear One, Unit No. 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated September 30, 2004, as supplemented by letters dated April 26 and June 8, 2005.

The amendment changes the existing steam generator (SG) tube surveillance program to be consistent with that being proposed by the TS Task Force (TSTF) in TSTF-449. These changes revise definitions in TS 1.1, reactor coolant system operational leakage in TS 3.4.13, SG program in TS 5.5.9, and SG tube inspection reports in TS 5.6.7, and add a new TS 3.4.16 on SG tube integrity. Also, as a result of the licensee replacing the SGs with SGs having a new Alloy 690 thermally treated tubing design, the TSs are revised to reflect this replacement.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/
Mohan C. Thadani, Senior Project Manager, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosures: 1. Amendment No. 224 to DPR-51
2. Safety Evaluation

cc w/encls: See next page

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DATE	8/9/05	8/9/05	07/07/05	04/12/05*	07/05/05

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DATE	07/06/05	8/9/05	8/10/05

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ENTERGY OPERATIONS INC.

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 224
Renewed License No. DPR-51

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Entergy Operations, Inc. (the licensee) dated September 30, 2004, as supplemented by letters dated April 26 and June 8, 2005, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment; and paragraph 2.c.(2) of Renewed Facility Operating License No. DPR-51 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 224, are hereby incorporated in the renewed license. EOI shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance and shall be implemented prior to resumption of operation from the 1R19 refueling outage scheduled for the fall of 2005.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

David Terao, Chief, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: August 10, 2005

ATTACHMENT TO LICENSE AMENDMENT NO. 224

RENEWED FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
1.1-3	1.1-3
3.4.13-1	3.4.13-1
3.4.13-2	3.4.13-2
- - -	3.4.16-1
- - -	3.4.16-2
5.0-10	5.0-10
5.0-11	5.0-11
5.0-12	5.0-12
5.0-13	- - -
5.0-14	- - -
5.0-15	- - -
5.0-16	- - -
5.0-17	- - -
5.0-18	- - -
5.0-19	- - -
50-29	5.0-29

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 224 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-51

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT NO. 1

DOCKET NO. 50-313

1.0 INTRODUCTION

By application dated September 30, 2004 (Agencywide Documents and Access Management System (ADAMS) Accession No. ML042800387), as supplemented by letters dated April 26 and June 8, 2005 (ADAMS Accession Nos. ML051300292 and ML051660294), Entergy Operations, Inc. (the licensee), requested changes to the Technical Specifications (TSs) for Arkansas Nuclear One, Unit No. 1 (ANO-1). The supplement dated June 8, 2005, provided additional information that clarified the application, did not expand the scope of the application as re-noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's renoticed proposed no significant hazards consideration determination as published in the *Federal Register* on May 24, 2005 (70 FR 29790).

The proposed changes would revise the existing steam generator (SG) tube surveillance program to be consistent with that being proposed by the TS Task Force (TSTF) in TSTF-449. These changes would revise definitions in TS 1.1, reactor coolant system operational leakage in TS 3.4.13, SG program in TS 5.5.9, and SG tube inspection reports in TS 5.6.7, and add a new TS 3.4.16 on SG tube integrity. Also, as a result of the licensee replacing the existing SGs with SGs having a new Alloy 690 thermally treated tubing design, the TSs would be revised to reflect this replacement.

This amendment request is the culmination of NRC and industry efforts since the mid-1990s to develop a programmatic, largely performance-based regulatory framework for ensuring SG tube integrity. The scope of the TS amendment request includes:

a. Revised TS 1.1, “Definition”
b. Revised TS 3.4.13 and TS Bases B 3.4.13, “RCS Operational LEAKAGE”
c. New TS 3.4.16 and new TS Bases B 3.4.16, “Steam Generator (SG) Tube Integrity”
d. Revised TS 5.5.9, “Steam Generator (SG) Program”
e. Revised TS 5.6.7, “Steam Generator Tube Inspection Reports”

The licensee stated that the reason for the proposed changes is twofold, the first being that ANO-1 will be replacing the existing SGs in refueling outage 1R19 (scheduled to commence in

the fall of 2005) with replacement SGs having thermally treated Alloy 690 tubes. Therefore, the portions of the SG TS that only apply to the original SGs are to be removed. Second, the licensee is modifying the TS to be consistent with the NRC/industry developed programmatic, performance-based regulatory framework mentioned above.

The proposed new TS 3.4.16, "Steam Generator (SG) Tube Integrity," in conjunction with the proposed revisions to administrative TS 5.5.9, “Steam Generator (SG) Program,” would establish a new programmatic, largely performance-based framework for ensuring SG tube integrity. Proposed TS Bases B 3.4.16 documents the licensee’s bases for this framework. Proposed TS 3.4.16 would establish new limiting conditions for operation (LCOs) related to SG tube integrity: namely, (1) SG tube integrity shall be maintained, and (2) all SG tubes exceeding the tube repair criteria (i.e., tubes with measured flaw sizes exceeding the tube repair criteria) shall be plugged in accordance with the SG Program. TS 3.4.16 would include surveillance requirements (SRs) to verify that the LCOs are met in accordance with the SG Program.

Proposed administrative TS 5.5.9, “Steam Generator (SG) Program,” would replace the current administrative TS 5.5.9, "Steam Generator Tube Surveillance Program." This revised TS would require establishing and implementing a program that ensures that SG tube integrity is maintained. SG tube integrity is defined in the proposed TS in terms of specified performance criteria for structural and leakage integrity. TS 5.5.9 would also provide for monitoring the condition of the tubes relative to these performance criteria during each SG tube inspection and for ensuring that SG tube integrity is maintained between scheduled inspections of the SG tubes. TS 5.5.9 would remove the SG-specific tube repair limits (i.e., alternate repair criteria) associated with the original SGs. The remaining tube repair limit would be retained. In addition, all references to SG tube sleeving are removed as the SG tube sleeving is not applicable to the replacement SGs.

The proposed changes to TS 5.6.7, “Steam Generator Tube Inspection Reports,” revise the existing requirements for, and the contents of, the SG tube inspection report consistent with the proposed revisions to TS 5.5.9. The current requirement for a 90-day report would be changed to a 180-day report.

The proposed amendment includes proposed revisions to TS 3.4.13 and its bases, “RCS Operational LEAKAGE.” The proposed amendment would retain the current LCO limit of 150 gallons per day (gpd) for primary-to-secondary leakage from any one SG. Retaining this latter requirement effectively ensures that total primary-to-secondary leakage through all the SGs is not allowed to exceed 300 gpd. (Note, ANO-1 is a two-loop plant.) The proposed changes would revise the TS 3.4.13 conditions and SRs to better clarify the requirements related to primary-to-secondary leakage.

Lastly, the proposed changes to TS 1.1, “Definition,” are editorial changes to the definition of LEAKAGE.

2.0 REGULATORY EVALUATION

2.1 Current Licensing Basis/SG Tube Integrity

The SG tubes in pressurized-water reactors (PWRs) have a number of important safety functions. These tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied upon to maintain primary system pressure and inventory. As part of the RCPB, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system and are relied upon to isolate the radioactive fission products in the primary coolant from the secondary system. In addition, the SG tubes are relied upon to maintain their integrity to be consistent with the containment objectives of preventing uncontrolled fission product release under conditions resulting from core damaging severe accidents.

Title 10 of the *Code of Federal Regulations* (10 CFR) establishes the fundamental regulatory requirements with respect to the integrity of the SG tubing. Specifically, the General Design Criteria (GDC) in Appendix A to 10 CFR Part 50 state that the RCPB shall have “an extremely low probability of abnormal leakage...and gross rupture" (GDC 14), "shall be designed with sufficient margin" (GDC 15 and 31), shall be of "the highest quality standards practical" (GDC 30), and shall be designed to permit "periodic inspection and testing...to assess...structural and leaktight integrity" (GDC 32). To this end, 10 CFR 50.55a specifies that components that are part of the RCPB must meet the requirements for Class 1 components in Section III of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Code). Section 50.55a further requires, in part, that throughout the service life of a PWR facility, ASME Code Class 1 components meet the requirements, except design and access provisions and pre-service examination requirements, in Section XI, "Rules for Inservice Inspection [ISI] of Nuclear Power Plant Components," of the ASME Code, to the extent practical. This requirement includes the inspection and repair criteria of Section XI of the ASME Code.

In the 1970s, Section XI requirements pertaining to ISI of SG tubing were augmented by additional SG tube SRs in the TSs. Paragraph (b)(2)(iii) of 10 CFR 50.55a states that where TS SRs for SGs differ from those in Article IWB-2000 of Section XI of the ASME Code, the ISI program shall be governed by the TSs.

The existing plant TSs include LCOs and accompanying SRs and action statements pertaining to the integrity of the SG tubing. SG operability in accordance with the SG tube surveillance program is necessary to satisfy the LCOs governing RCS loop operability, as stated in the accompanying TS Bases. The LCO governing RCS Operational LEAKAGE includes limits on allowable primary-to-secondary LEAKAGE through the SG tubing. Accompanying SRs require verification that RCS operational LEAKAGE is within limits every 72 hours by an RCS water inventory balance and that SG tube integrity is in accordance with the SG tube surveillance program. The SG tube surveillance program requirements are contained in the administrative TSs. These administrative TSs state that the SGs are to be determined OPERABLE after the actions required by the surveillance program are completed.

Under the plant TS SG surveillance program requirements, licensees are required to monitor the condition of the SG tubing and to perform repairs, as necessary. Specifically, licensees are required by the plant TSs to perform periodic ISIs and to remove from service, by plugging or

repair, all tubes found to contain flaws with sizes exceeding the acceptance limit, termed "plugging limit" (old terminology) or "tube repair criteria" (new terminology). The frequency and scope of the inspection and the tube repair limits are specified in the plant TSs.

The tube repair limits in the TSs were developed with the intent of ensuring that degraded tubes (1) maintain factors of safety against gross rupture consistent with the plant design basis (i.e., consistent with the stress limits of the ASME Code, Section III) and (2) maintain leakage integrity consistent with the plant licensing basis while, at the same time, allowing for potential flaw size measurement error and flaw growth between SG inspections.

As part of the plant licensing basis, applicants for PWR licenses are required to analyze the consequences of postulated design-basis accidents (DBAs) such as an SG tube rupture (SGTR) and main steamline break (MSLB). These analyses consider the primary-to-secondary leakage through the tubing which may occur during these events and must show that the offsite radiological consequences do not exceed the applicable guidelines of 10 CFR 100.11 (as supplemented by accident-specific criteria in Section 15 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants") for offsite doses, GDC-19 criteria (as supplemented by Section 6.4 of NUREG-0800) for control room operator doses, or some fraction thereof as appropriate to the accident, or the NRC-approved licensing basis (e.g., a small fraction of these limits).

2.2 <u>10 CFR 50.36</u>

In 10 CFR 50.36, the Commission established its regulatory requirements related to the content of TSs. In doing so, the Commission emphasized those matters related to the preventing of accidents and mitigating their consequences. As recorded in the Statements of Consideration, Technical Specifications for Facility Licenses: Safety Analysis Reports (33 FR 18610, December 17, 1968), the Commission noted that applicants are expected to incorporate into their TSs those items that are directly related to maintaining the integrity of the physical barriers designed to contain radioactivity. Pursuant to 10 CFR 50.36, TSs are required to include items in five specific categories related to station operation. Specifically, those categories include: (1) safety limits, limiting safety system settings, and limiting control settings; (2) LCOs; (3) SRs; (4) design features; and (5) administrative controls. However, the rule does not specify the particular requirements to be included in a plant's TS. The licensee’s September 30, 2004, application contains proposed LCOs, SRs, and administrative controls involving SG integrity, an important element of the physical barriers designed to contain radioactivity.

Additionally, 10 CFR 50.36(c)(2)(ii) sets forth four criteria to be used in determining whether an LCO is required to be included in the TS for a certain item. These criteria are as follows:

1. Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the RCPB.

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

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

The NRC staff has reviewed the proposed changes to ensure that these changes conform with 10 CFR 50.36 as discussed herein.

2.3 Background - TS Amendment Request

The current TS requirements for inspection and repair of SG tubing date to the mid-1970s and define a prescriptive approach for ensuring SG tube integrity. This prescriptive approach involves inspection of the tubing at specified intervals, implementation of specified tube inspection sampling plans, and repair or removal from service by plugging all tubes found by inspection to contain flaws in excess of specified flaw repair criteria. However, as evidenced by operating experience, the prescriptive approach defined in the TSs is not sufficient in-and-of-itself to ensure that SG tube integrity is maintained. For example, in cases of low to moderate levels of degradation, the TSs only require that 3 to 21 percent of the tubes be inspected, irrespective of whether the inspection results indicate that additional tubes may need to be inspected to reasonably ensure that tubes with flaws that may exceed the tube repair criteria, or that may impair SG tube integrity are detected. In addition, the TSs (and ASME Code, Section XI) do not explicitly address the inspection methods to be employed for different tube degradation mechanisms or tube locations, nor are the specific objectives to be fulfilled by the selected methods explicitly defined. Also, incremental flaw growth between inspections can, in many instances, exceed what is allowed in the specified tube repair criteria. In such cases, the specified inspection frequencies may not ensure reinspection of a tube before its integrity is impaired. In short, the current TS SRs do not require licensees to actively manage its SG surveillance programs so as to provide reasonable assurance that SG tube integrity is maintained.

In view of the shortcomings of the current TS requirements, licensees experiencing significant degradation problems have frequently found it necessary to implement measures beyond minimum TS requirements to ensure that adequate SG tube integrity is being maintained. Until the 1990s, these measures tended to be ad hoc. By letter dated December 16, 1997 (Reference 1), the Nuclear Energy Institute (NEI) provided NRC with a copy of NEI 97-06 (Original), “Steam Generator Program Guidelines,” and informed the NRC of the following formal industry position.

> Each licensee will evaluate its existing steam generator program and, where necessary, revise and strengthen program attributes to meet the intent of the guidance provided in NEI 97-06, “Steam Generator Program Guidelines,” no later than the first refueling outage starting after January 1, 1999.

The stated objectives of this initiative were to have a clear commitment from utility executives to follow industry SG related guidelines developed through the Electric Power Research Institute (EPRI) to assure a unified industry approach to emerging SG issues and to apply SG tube integrity performance criteria in conjunction with the performance-based philosophy of the

maintenance rule, 10 CFR 50.65. Reference 2 is the most recent update to NEI 97-06 available to the NRC staff. NEI 97-06 provides general, high-level guidelines for a programmatic, performance- based approach to ensuring SG tube integrity. NEI 97-06 references a number of detailed EPRI guideline documents for programmatic details. Subsequently, the NRC staff had extensive interaction with the industry to resolve NRC staff concerns with this industry initiative and to identify needed changes to the plant TSs to ensure that SG tube integrity is maintained (Reference 3).

Ultimately, in consideration of the performance-based objective of this initiative, the NRC staff determined it was not necessary for the NRC staff to formally review or endorse the NEI 97-06 guidelines or the EPRI guideline documents referenced by NEI 97-06. The NRC staff reviewed and approved amended TSs for Farley Units 1 and 2, which are programmatically consistent with the industry's NEI 97-06 initiative and which ensure that the licensee will implement an SG program that provides reasonable assurance that SG tube integrity will be maintained (Reference 4). The TS amendment being requested for ANO-1 is very similar to the Farley amendment.

3.0 TECHNICAL EVALUATION

3.1 TS 3.4.16, "Steam Generator (SG) Tube Integrity"

The current TS establishes an operability requirement for the SG tubing; namely, the tubes shall be determined OPERABLE after completion of the actions defined in the SG tube surveillance program (TS 5.5.9). In addition, this surveillance program (and SG operability) is directly invoked by TS 3.4.13 which contains the LCO relating to RCS leakage. However, these TSs do not directly require that SG tube integrity be maintained. Instead, they require implementation of an SG tube surveillance program which is assumed to ensure SG tube integrity, but, as discussed above, may not ensure SG tube integrity depending on the circumstances of degradation at a plant.

To address this shortcoming, the ANO-1 amendment application includes a proposed new specification, TS 3.4.16, "Steam Generator (SG) Tube Integrity," which includes a new LCO requirement and accompanying conditions, required actions, completion times, and SRs. The new LCO is applicable in MODES 1, 2, 3, and 4 and requires: 1) SG tube integrity shall be maintained, AND 2) all SG tubes satisfying the tube repair criteria shall be plugged in accordance with the SG Program (specified in the proposed TS 5.5.9). This LCO supplements the LCO in TS 3.4.13 to directly make SG tube integrity an operating restriction. This is consistent with Criterion 2 of 10 CFR 50.36(c)(2)(ii) since the assumption of SG tube integrity as an initial condition is implicit in DBA analyses (with the exception of analysis of a design-basis SG tube rupture (SGTR) where one tube is assumed not to have structural integrity), and is, therefore, acceptable to the NRC staff.

Proposed SR 3.4.16.1 would require that SG tube integrity be verified in accordance with the SG Program, which is described in proposed revisions to TS 5.5.9. The required frequency for this surveillance would also be in accordance with the SG Program, thus meeting the requirements of 10 CFR 50.36(c)(3). The revised TS 5.5.9 would define SG tube integrity in terms of satisfying SG tube integrity performance criteria for tube structural integrity and leakage integrity as specified therein. SR 3.4.16.1 would replace the existing SR (SR 3.4.13.2) in the RCS Operational LEAKAGE TS (TS 3.4.13), which provides that SG tube integrity be

verified in accordance with the SG surveillance program as provided in the current TS 5.5.9. The proposed SR improves upon the current SR in that it refers to a program that is directly focused on maintaining SG tube integrity rather than on implementing a prescriptive surveillance program that, as discussed above, may not be sufficient to ensure SG tube integrity is maintained. Proposed SR 3.4.16.2 would require verification that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the SG Program. The tube repair criteria are contained in the SG Program. The required frequency for SR 3.4.16.2 is prior to entering MODE 4 following a SG tube inspection. The NRC staff concludes that the proposed SR 3.4.16.1 and SR 3.4.16.2 are sufficient to determine whether the proposed LCO is met, meet the requirements of 10 CFR 50.36(c)(3), and are acceptable.

The licensee has proposed conditions, required actions, and completion times for the new LCO 3.4.16 as shown in Table 1. The proposed TS 3.4.16 allows separate condition entry for each SG tube.

Table 1 - TS 3.4.16 ACTIONS

Condition	Required Action	Completion Time
A. One or more SG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next inspection. <u>AND</u> A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	7 days Prior to entering MODE 4 following the next refueling outage or SG tube inspection.
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> SG tube integrity not maintained.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5	6 hours 36 hours

Should SG tube integrity be found by the SG Program not to be maintained, Required Actions B.1 and B.2 would require that the plant be in MODE 3 within 6 hours and MODE 5 within 36 hours, respectively. These required actions and completion times are consistent with (1) the general requirements in TS 3.0.3 for failing to meet an LCO and (2) the requirements of TS 3.4.13 when the LCO on primary-to-secondary leakage rate is not met. The NRC staff concludes that these required actions and completion times provide adequate remedial measures should SG tube integrity be found not to be maintained and are acceptable to the NRC staff.

Condition A of proposed TS 3.4.16 addresses the condition where one or more tubes satisfying the tube repair criteria are inadvertently not plugged in accordance with the SG Program. Under Required Action A.1, the licensee would be required to verify within 7 days that SG tube integrity of the affected tubes is maintained until the next inspection. The accompanying Bases state that the SG tube integrity determination would be based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next inspection. The NRC staff notes that details of how this assessment would be performed are not included in proposed TS 3.4.16 or 5.5.9. The NRC staff finds this to be consistent with having performance-based requirements, finds that the performance criteria (i.e., performance objectives) for assessing SG tube integrity are clearly defined (in TS 5.5.9), and finds that it is appropriate that the licensee have the flexibility to determine how best to perform this assessment based on what information is and is not available concerning the circumstances of the subject flaw. The proposed 7 days allowed to complete the assessment ensures that the risk increment associated with operating with tubes in this condition will be very small. Should the assessment reveal that SG tube integrity cannot be maintained until the next scheduled inspection or if the assessment is not completed in 7 days, Condition B applies, leading to Required Actions B.1 and B.2, which are evaluated above. Finally, if Required Action A.1 successfully verifies that SG tube integrity is being maintained until the next inspection, Required Action A.2 would require that the subject tube be plugged in accordance with the SG Program prior to entering MODE 4 after the next refueling outage or SG inspection. Based on the above, the NRC staff concludes that the proposed LCO and accompanying ACTIONS related to failure to plug a tube that satisfies the tube repair criteria are acceptable.

The licensee has proposed changes to the Bases supporting the proposed new TS 3.4.16. Although the TS Bases are controlled under the provisions of 10 CFR 50.59 and TS 5.5.14, TS Bases Control Program; the NRC staff finds the proposed changes to the proposed TS 3.4.16 Bases to be acceptable.

3.2 SG Operability

TS Bases for B3.4.5 defines the RCS Loops-Mode 3 as an OPERABLE RCS loops consisting of at least one OPERABLE RCP and an SG that is OPERABLE. To be considered OPERABLE RCP must be capable of being powered and able to provide forced flow if required. Similarly, SG must be capable of transferring heat from the reactor coolant at a controlled rate and be in compliance with the Steam Generator Tube Surveillance Program. The TS Bases B3.4.6, RCS Loops_MODE 4, defines the OPERABLE RCS loop as consisting of at lease one OPERABLE RCP and an OPERABLE SG. To be considered OPERABLE, an SG must be capable of transferring heat from the reactor coolant at a controlled rate and be in compliance with the Steam Generator Tube Surveillance Program. In the Bases B3.4.7 for TS 3.4.7, RCS Loops-Mode 5, Loops Filled, defines the OPERABLE SG as an SG that can perform as a heat sink when it has an adequate water level and is in compliance with Steam Generator Tube Surveillance Program. The TS Bases are controlled under the provisions of 10 CFR 50.59 and TS 5.5.14, TS Bases Control Program. The licensee has proposed to delete the phrases, “in accordance with the Steam Generator Tube Surveillance Program,” from these Bases.

With the deletion of these phrases, an OPERABLE SG will be defined under the definition of OPERABLE - OPERABILITY defined in TS 1.1 and as stated below:

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

The NRC staff has evaluated the proposed Bases changes. The current Bases refer to the SG Tube Surveillance Program for the requirements of an OPERABLE SG. The SG Tube Surveillance Program provided the controls for the ISI of SG tubes that was intended to ensure that the structural and leakage integrity of this portion of the RCS was maintained. Using the definition of OPERABLE - OPERABILITY expands the definition of an OPERABLE SG beyond maintaining structural and leakage integrity and is acceptable. Therefore, the NRC staff finds the proposed changes to the Bases for TSs 3.4.5, 3.4.6, and 3.4.7 to be acceptable.

3.3 Proposed Administrative TS 5.5.9, "Steam Generator (SG) Program"

The proposed administrative TS 5.5.9, "Steam Generator Program," replaces the existing administrative TS 5.5.9, “Steam Generator Tube Surveillance Program.” The current TS 5.5.9 defines a prescriptive strategy for ensuring SG tube integrity consisting of tube inspections performed at specified intervals, with a specified inspection scope (tube inspection sample sizes), and with a specified tube acceptance limit for degraded tubing, termed “tube repair criterion,” beyond which the affected tubes must be plugged or repaired. In addition, the current TS 5.5.9 defines acceptable tube repair limits, some of which are only applicable to the original SGs. The proposed TS 5.5.9 incorporates a largely performance-based strategy for ensuring SG tube integrity, requiring that a SG Program be established and implemented to ensure SG tube integrity is maintained. The proposed specification contains only a few details concerning how this is to be accomplished, the intent being that the licensee will have the flexibility to determine the specific strategy to be employed to satisfy the required objective of maintaining SG tube integrity. However, as evaluated below, the NRC staff concludes that the proposed TS 5.5.9 provides reasonable assurance that the SG Program will maintain SG tube integrity.

The proposed Bases for TS 3.4.16 state that NEI 97-06 and its referenced EPRI guideline documents will be used to establish the content of the SG Program. The guidelines are industry-controlled documents and licensee SG programs may deviate from these guidelines. Except as may be specifically invoked by the TSs, the NRC staff’s evaluation herein takes no credit for any of the specifics in the guidelines.

3.3.1 Performance Criteria for SG Tube Integrity

Proposed TS 5.5.9 would require that SG tube integrity be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage as specified therein.

The NRC staff’s criteria for evaluating the acceptability of these performance criteria are that meeting these criteria are sufficient to ensure that SG tube integrity is within the plant licensing basis and that meeting these criteria, in conjunction with implementation of the SG Program,

ensures no significant increase in risk. These performance criteria must also be evaluated in the context of the overall SG Program such that if the performance criteria are inadvertently exceeded, the consequences will be tolerable before the situation is identified and corrected. In addition, the performance criteria must be expressed in terms of parameters that are measurable, directly or indirectly.

3.3.1.1 Structural Integrity Criterion

The proposed structural integrity criterion is as follows:

> All inservice 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 design basis accident primary to secondary pressure differentials. Apart from the above requirements, additional loading considerations associated with 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.

The NRC staff has evaluated this proposed criterion for consistency with the safety factors embodied in the current licensing basis, specifically, the safety factors embodied in the TS tube repair criteria. The tube repair criterion typically specified in plant TSs is 40 percent of the initial tube wall thickness. This criterion is typically applicable to all tubing flaws found by inspection, except for certain flaw types at certain locations for which less restrictive repair criteria may be applicable (as specified in the TSs) and for certain sleeve repairs for which a more restrictive tube repair criterion may be specified. For ANO-1, the 40 percent tube repair criterion is the only such criterion that will be applicable to the replacement SGs and is applicable to all flaw types at all tube locations. The less restrictive repair criteria and sleeve repairs in the current TS will be eliminated as part of the proposed change.

In 1976 the NRC staff prepared Regulatory Guide (RG) 1.121 (Draft), “Basis for Plugging Degraded PWR Steam Generator Tubes,” (Reference 5) describing a technical basis for the development of tube repair criteria. This draft RG was issued for public comment, but was never finalized. Although not finalized, the RG is generally cited in licensee and industry documentation as the basis for the TS tube repair criteria in plant TSs. The draft RG includes the following with respect to safety factors:

d. Degraded tubing should retain a factor of safety against burst of not less than three under normal operating conditions.

e. Degraded tubing should not be stressed beyond the elastic range of the tube material during the full range of normal reactor operation. The draft RG also states that loadings

associated with normal plant conditions, including startup, operation in the power range, hot standby, and cooldown, as well as all anticipated transients (e.g., loss of electrical load, loss of off-site power) that are included in the design specifications for the plant, should not produce a primary membrane stress in excess of the yield stress of the tube material at operating temperature.

f. Degraded tubes should maintain a margin of safety against tube failure under postulated accidents consistent with the margin of safety determined by the stress limits specified in NB-3225 of Section III of the ASME Code. Note, NB-3225 specifies that the rules in Appendix F of Section III may be used for evaluating these loadings.

The “safety factor of three” criterion stems from Section III of the ASME Code which, in part, limits primary membrane stress under design conditions to one third of ultimate strength. The proposed structural integrity criterion would limit application of the “safety factor of three” criterion to only those pressure loadings existing during normal full power, steady state operating conditions. Differential pressures under this condition are plant-specific, ranging from 1250 psi to 1500 psi (Reference 6). However, differential pressure loadings can be considerably higher during normal operating transients, ranging to between 1600 psi to 2150 psi during plant heatup and cooldown transients (Reference 6). Given a factor of safety equal to three under normal full power conditions, the factor of safety during heatups and cooldowns can be as low as about two. The industry stated in a white paper (Reference 6) that it was not the intent of the 40 percent depth-based tube repair criterion to ensure a factor of safety of three for operating transients such as heatups and cooldowns. The industry stated that maintaining a safety factor of three for such transients would lead to tube repair criteria less than the standard 40 percent criterion for many plants. The NRC staff has independently performed calculations that support the industry’s contention that applying the “safety factor of three” criterion to the full range of normal operating conditions would lead to tube repair criteria more restrictive than the 40 percent criterion, which the NRC staff has accepted since the 1970s. The NRC staff concludes that the “safety factor of three” criterion for application to normal full power, steady state pressure differentials, as proposed by the licensee and the industry, is consistent with the safety margins implicit in existing TS tube repair criteria and, thus, is consistent with the current licensing basis.

Item b above from draft RG 1.121 is often referred to as the “no yield” criterion. The purpose of this criterion is to prevent permanent deformation of the tube to assure that degradation of the tube will not occur due to mechanical effects of the service condition. This is consistent with ASME Code, Section III, stress limits, which serve to limit primary membrane stress to less than yield. The proposed structural integrity criteria do not include this “no yield” criterion. The industry states in its white paper (Reference 6) that, if a tube satisfies the “safety factor of three” criterion at full power operating pressure differentials, the tube will generally satisfy the “no yield” criterion for the operating transient (e.g., heatup and cooldown) pressure differentials. The white paper acknowledges that this may not be true for all plant-specific conditions and material properties. For this reason, NEI 97-06, Rev. 1, and the EPRI Steam Generator Integrity Assessment Guidelines state that, in addition to meeting the safety factor of three for normal steady state operation, the integrity evaluation shall verify that the primary pressure stresses do not exceed the yield strength for the full range of normal operating conditions. The white paper, which has been incorporated as part of the EPRI Steam Generator Integrity Assessment Guidelines, recommends that this be demonstrated for each plant using plant-specific conditions and material properties.

The NRC staff concurs that the “no yield” criterion need not be specifically spelled out in the TS definition of the structural integrity criteria. The NRC staff finds that the appropriate focus of the TS criteria should be on preventing burst. The NRC staff calculations confirm that the proposed “safety factor of three” criterion bounds or comes close to bounding the “no yield” criterion for most of the cases investigated. This is not absolute, however. For once-through SGs (OTSGs), the NRC staff noted a case where elastic hoop stress in a uniformly thinned tube could exceed the yield strength by 20 percent under heatup and cooldown conditions and still satisfy the “safety factor of three” criterion against burst under normal steady state, full power operating conditions. Such a tube would still retain a factor of safety of two against burst under heatup and cooldown conditions. The amount of plastic strain induced would be limited to between 1 and 2 percent based on typical strain hardening characteristics of the material. This is quite small compared to cold working associated with fabrication of tube u-bends and tube expansions. Operating experience shows that this level of plastic strain (i.e., permanent strain caused by exceeding the yield stress) has not adversely affected the stress corrosion cracking resistance of OTSG tubing relative to that expected for non-plastically strained tubing. Thus, the NRC staff concludes that the “safety factor of three” criterion is sufficient to limit plastic strains to values that will not contribute significantly to degradation of the tubing and that the “no yield’ criterion need not be specifically spelled out in the structural integrity performance criteria.

The proposed safety factor of 1.4 against burst applied to design-basis primary-to-secondary pressure differentials derives from the 0.7 times ultimate strength limit for primary membrane stress in the ASME Code, Appendix F, F-1331.1(a). This criterion is consistent with the stress limit criteria used to develop the standard 40 percent tube repair criteria in the TSs and with the safety factor criteria used in the derivation of alternate tube repair criteria in plant TSs, such as the voltage based criteria for outer-diameter stress corrosion cracking. Thus, the criterion is consistent with the current licensing basis and is acceptable.

Apart from differential pressure loadings, other types of loads may also contribute to burst. Examples of such loads include bending moments on the tubes due to flow induced vibration, earthquake, and loss-of-coolant accident (LOCA) rarefaction waves. For OTSGs, axial loads are induced in the tubes due to pressure loadings acting on the SG shell and tube sheets and due to differential thermal expansion between the tubes and the SG shell. Such non-pressure loads generally produce negligible primary stress during normal operating conditions from the standpoint of influencing burst pressure. In general, such non-pressure loads may be more significant under certain accident loadings depending on SG design, flaw location, and flaw orientation. Such non-pressure sources of primary stress under accident conditions were explicitly considered in the development of the 40 percent tube repair criterion relative to ASME Code, Appendix F, stress limits.

The proposed structural criterion requires that, apart from the safety-factor requirements applying to pressure loads, additional loads associated with DBAs, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine whether these loads contribute significantly to burst or collapse. The NRC staff notes that examples of such additional loads include bending moments during LOCA, MSLB, or safe shutdown earthquake (SSE) and axial, differential thermal loads. “Combination of accidents” refers to the fact that the design and licensing basis for many plants is that DBAs, such as LOCA and MSLB, are assumed to occur concurrently with SSE. Whereas “burst” is the failure mode of interest where primary-to-secondary pressure loads are dominant, “collapse” is a potential limiting

failure mode (although an unlikely one, according to industry based on a recent study (Reference 7)) for loads other than pressure loads. “Collapse” refers to the condition where the tube is not capable of resisting further applied loading without unlimited displacement. Although the occurrence of a collapsed tube or tubes would not necessarily lead to perforation of the tube wall, the consequences of tube collapse have not been analyzed and, thus, the NRC staff finds it both appropriate and conservative to ensure there is margin relative to such a condition.

Where non-pressure loads are determined to significantly contribute to burst or collapse, the proposed structural criteria require that such loads 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 a 1.0 safety factor on axial secondary loads. The 1.2 safety factor for combined primary loads was derived from the ratio of burst or collapse load divided by allowable load from ASME Code for faulted conditions. Burst or collapse load was assumed to be equal to the material flow stress, assuming Code minimum yield and ultimate strength values and a flow stress coefficient of 0.5. Allowable load was determined from ASME Code, Section III, Appendix F, F-1331.3.a, which defines an allowable primary membrane plus bending load for service level d (faulted) conditions. The NRC staff finds this 1.2 safety factor acceptable. The proposed 1.0 safety factor for axial secondary loads goes beyond what is required by the design basis in Section III of the ASME Code, since Section III assumes that a one-time application of such a load cannot lead to burst or collapse. However, this is not necessarily the case for tubes with circumferential cracks. The proposed safety factor criterion of 1.0 is conservative for loads that behave as secondary since it ignores the load relaxation effect associated with axial yielding before tube severance (burst) occurs.

It is the NRC staff's position that, in accordance with 10 CFR 50.46, the spectrum of DBAs should include large-break LOCAs (LBLOCAs). Inherent in the evaluations performed in support of 10 CFR 50.46 are assumptions regarding SG tube integrity (e.g., the tubes remain intact). Under the current TSs, the SGs are considered operable provided the SG tube surveillance requirements in TS 5.5.9 are met, irrespective of the treatment of LBLOCA loads when performing SG tube integrity assessments. However, under the proposed new SG TS, the finding of a flaw such as a large circumferential indication would raise an operability issue unless the licensee could demonstrate that the tubes in which the circumferential indications are found continue to meet the safety factors in the structural integrity performance criteria under LBLOCA conditions. Large flaws which could lead to tube failures during a LBLOCA are not likely to occur early in the life of the ANO-1 replacement SGs (if ever at all) based on operating experience to date with SGs employing Alloy 690 thermally treated (TT) tubing. In the meantime, the Babcock & Wilcox Owners Group is currently pursuing a topical report initiative (known as BAW-2374 regarding SG tube loads following a LBLOCA) to develop a technical basis for redefining SG tube integrity during a LBLOCA for nuclear plants designed by Babcock and Wilcox.

Apart from being consistent with the current licensing basis, NRC risk studies have indicated that maintaining the performance criteria safety factors is important to avoiding undue risk, particularly risk associated with severe accident scenarios involving a fully pressurized primary system and depressurized secondary system and where the tubes may heat to temperatures well above design basis values, significantly reducing the strength of the tubes (Reference 8).

Based on the above, the NRC staff finds that the proposed structural performance criterion is consistent with the margins of safety embodied in existing plant licensing bases. Exceeding

this criterion is not likely to lead to consequences that are intolerable provided that such an occurrence is infrequent and that, if it occurs, it is promptly detected and corrected so as to ensure that risk is limited. Even if a tube should degrade to the point of rupture under normal operating conditions, such an occurrence is an analyzed condition with reasonable assurance that the radiological consequences will be acceptable. Finally, the structural performance criterion is expressed in terms of parameters that are measurable. Specifically, structural margins can be directly demonstrated through in situ pressure testing or can be calculated from burst prediction models using as input flaw size measurements obtained by inspection. Thus, the NRC staff finds the proposed structural performance criterion to be acceptable.

3.3.1.2 Accident Leakage Criterion

The proposed accident induced leak rate criteria is as follows:

> 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.

This performance criterion for accident induced leak rate is consistent with leak rates assumed in the licensing basis accident analyses for purposes of demonstrating that the consequences of DBAs meet the dose guidelines in 10 CFR Part 100 for offsite doses, GDC 19 for control room operator doses, or some fraction thereof as appropriate to the accident, or the NRC-approved licensing basis (e.g., a small fraction of these dose guidelines). This criterion does not apply to design basis SGTR accidents for which leakage corresponding to a postulated double ended rupture of a tube is assumed in the analysis. The proposed criterion ensures that from the standpoint of accident induced leakage the plant will be operated within its analyzed condition and is acceptable.

For certain severe accident sequences involving high primary side pressure and a depressurized secondary system (“high-dry” condition), primary-to-secondary leakage may lead to more heating of the leaking tube than would be the case were it not leaking, thus increasing the potential for failure of that tube and a consequent large early release. The proposed 1.0 gpm limit on total leakage during DBAs (other than a SGTR) ensures that the potential for induced leakage during severe accidents will be maintained at a level that will not increase risk.

Exceeding this criterion is not likely to lead to intolerable consequences provided that such an occurrence is infrequent and that such an occurrence, if it occurs, is promptly detected and corrected so as to ensure that risk is minimized. It should be noted that the criterion applies to leakage that could be induced by an accident in the unlikely event that such an accident occurs. Finally, the accident leakage performance criterion is expressed in terms of parameters that are measurable, both directly and indirectly. Specifically, structural margins can be directly demonstrated through in situ pressure testing or can be calculated using leakage prediction models using flaw size measurements obtained by ISI as input.

Based on the foregoing, the NRC staff finds the proposed accident leakage performance criterion to be acceptable.

3.3.1.3 Operational Leakage Criterion

Proposed TS 5.5.9 states that the operational leakage performance criterion is specified in LCO 3.4.13, “RCS Operational LEAKAGE.” Given the TS LCO limit, a separate performance criterion for operational leakage is unnecessary for ensuring prompt shutdown should the limit be exceeded. However, operational leakage is an indicator of SG tube integrity performance, though not a direct indicator. It is the only indicator that can be monitored while the plant is operating. Maintaining leakage to within the limit provides added assurance that the structural and accident leakage performance criteria are being met. Thus, the NRC staff believes that inclusion of the TS leakage limit among the set of SG tube integrity performance criteria is appropriate from the standpoint of completeness and is, therefore, acceptable.

3.3.2 Condition Monitoring Assessment

Proposed TS 5.5.9 would require that the SG Program include provisions for condition monitoring assessments as follows:

> 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 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.

The NRC staff finds that the proposed requirement for condition monitoring assessments addresses an essential element of any performance-based strategy, namely, the need to monitor performance relative to the performance criteria. Confirmation that the SG tube integrity criteria are met would confirm that the overall programmatic goal of maintaining SG tube integrity has been met to that point in time. However, failure to meet the SG tube integrity criteria would be indicative of potential shortcomings in the effectiveness of the licensee’s SG Program and the need for corrective actions relative to the program to ensure that SG tube integrity is maintained in the future. Failure to meet either the structural or accident leakage performance criterion would be reportable pursuant to 10 CFR 50.72 and 50.73 in accordance with guidelines in Reference 9. In addition, the NRC Regional Office would follow up on such an occurrence as appropriate consistent with the NRC Reactor Oversight Process (ROP) (Reference 10) and the risk significance of the occurrence.

TS 5.5.9 would require that condition monitoring be performed at each ISI of the tubing. The NRC staff’s evaluation of the proposed frequency of ISI is addressed in Section 3.3.3 of this safety evaluation.

3.3.3 Inservice Inspection

The proposed TS 5.5.9 would require that the SG Program include periodic tube inspections. This proposal includes a new performance-based requirement that the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next inspection. This is a performance-based requirement that

complements the requirement for condition monitoring from the standpoint of ensuring SG tube integrity is maintained. The requirement for condition monitoring is backward looking in that it is intended to confirm that SG tube integrity has been maintained up to the time the assessment is performed. The ISI requirement, by contrast, is forward looking. It is intended to ensure that tube inspections in conjunction with plugging of tubes are performed such as to ensure that the performance criteria will continue to be met at the next SG inspection. This would be followed again by condition monitoring at the next SG inspection to confirm that the performance criteria were in fact met.

With respect to scope and methods of inspection, the proposed specification would also require that the number and portions of tubes inspected and method of inspection be performed with the objective of detecting flaws of any type (for example, 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. Furthermore, 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.

The NRC staff finds that this proposal concerning the scope and methods of inspection includes a number of improvements relative to the current specification. The proposed requirement clarifies the licensee's obligation under existing TSs and 10 CFR Part 50, Appendix B, to employ inspection methods capable of detecting flaws of any type that the licensee believes may potentially be present anywhere along the length of the tube based on a degradation assessment.

The proposed specification specifically excludes the tubesheet welds and the tube ends beyond the welds from the inspection requirements therein. The NRC staff finds this to be consistent with current actual practice and to be acceptable. The tube ends beyond the tube-to-tubesheet welds are not part of the primary pressure boundary.

The proposed specification would replace current specific requirements pertaining to the number of tubes to be inspected at each inspection, in part, with a requirement that is performance based; that is, the number and portions of tubes inspected (in conjunction with other elements of inspection) shall be such as to ensure that SG tube integrity is maintained until the next inspection. The current minimum tube sampling requirement for an SG inspection is 3 percent of the SG tubing at the plant. The purpose of this initial sample is to determine whether active degradation is present and whether there is a need to perform additional inspection sampling. Actual industry practice, consistent with NEI 97-06 and the EPRI Examination Guidelines, Rev. 6, typically involves initial inspection samples of at least 20 percent. If moderate numbers of tubes (i.e., category C-2 as defined in the current TS) are found to contain flaws, the current TS requires that an additional 6 to 18 percent of the tubes be inspected. In many cases this requirement is very non-conservative since no consideration is given to whether uninspected tubes may contain flaws that could challenge the SG tube integrity performance criteria prior to the next inspection. Current industry practice and the industry guidelines involve substantially higher levels of sampling under these circumstances. This practice has been motivated by a desire to minimize forced outages as well as by the requirement to ensure SG tube integrity. The NRC staff finds, therefore, that current TS sampling requirements do not drive actual sampling programs in the field for plants with low to

moderate levels of tube degradation, and that for moderate levels of tube degradation the current TS requirements do not ensure adequate levels of sampling to ensure SG tube integrity will be maintained. The proposed specification addresses this shortcoming by requiring that inspection scope be consistent with the overall performance objective that SG tube integrity be maintained until the next SG inspection.

For SGs with high levels of degradation (i.e., category C-3 as defined in current TS), the current TS requires that the inspections be expanded to include 100 percent of the tubes in the affected SG. This requirement is conservative in cases where the active degradation is confined to specific groups of tubes in the SG. This requirement does drive actual sampling programs in the field since industry guidelines would permit 100 percent sampling to be confined to those portions of the SG bounding the region where the degradation has been found to be active. The proposed specification would give licensees the flexibility to implement less than 100 percent inspection of the SG tubes in these cases provided it is consistent with the performance-based objective of ensuring that SG tube integrity is maintained until the next SG inspection.

Overall, the NRC staff concludes that the proposed specification ensures that the licensee will implement inspection scopes consistent with the overall objective that SG tube integrity be maintained. To meet this requirement, it will be necessary to inspect tubes that may contain flaws that may challenge the SG tube integrity performance criteria prior to the next inspection. The proposed specification gives the licensee the flexibility to define an inspection scope that ensures that this objective is met while avoiding any unnecessary inspections.

With respect to frequency of inspection, the current specification requires that SG inspections be performed every 24 calendar months. This frequency may be extended to once every 40 calendar months if the previous two inspections revealed only low-level degradation (i.e., category C-1 results as defined in the TS). The inspection frequency is required to revert from the 40 calendar months to 20 calendar months if an extensive level of degradation (i.e., category C-3 results as defined in the TS) was observed during the most recent inspection. Except in cases where extensive degradation (i.e., category C-3) is found in any SG, SGs may be inspected on a rotating basis at each inspection. Thus, for 2-loop plants performing SG inspections at 24-month intervals, intervals for individual SGs may range to 48 months. Similarly, for 2-loop plants performing SG inspections at 40-month intervals, intervals for individual SGs may range to 80 months. However, these prescriptive requirements bear no direct relationship to the overall objective of ensuring that SG tube integrity is maintained. These requirements apply irrespective of the flaw detection and sizing performance of the inspection methods utilized and the rate at which flaws may be growing in the subject SGs. These requirements do not ensure that flawed tubing remaining in service following an SG tube inspection and the incremental flaw growth that may take place prior to the next inspection are within the allowances provided for by the TS tube repair limit or that SG tube integrity will be maintained prior to the next inspection.

Plants operating with their originally installed SGs have typically inspected each SG at each refueling outage, which typically occur at intervals of less than 24 calendar months. The vast majority of these SGs contained Alloy 600 mill annealed (MA) tubing, which quickly became moderately to extensively degraded (i.e., category C-2 or C-3 as defined in the current TS) such that the TS would not allow longer intervals. The 24-month inspection interval requirement usually proved sufficient in maintaining SG tube integrity. Nonetheless, there have been

instances where licensees have performed mid-cycle inspections to ensure SG tube integrity would be maintained.

However, many SGs with Alloy 600 MA tubing have been replaced with SGs with Alloy 600 TT or Alloy 690 TT tubing, which have proven to be much more resistant to stress corrosion cracking (SCC) than Alloy 600 MA tubing. This includes ANO-1, which is replacing SGs in the fall 2005 with SGs using Alloy 690 TT tubing. Based on early low levels of degradation, some of the plants with replacement SGs are taking advantage of the longer inspection intervals permitted by the TSs.

Under the proposed TS 5.5.9, the required frequency of inspection in conjunction with inspection scope and inspection methods shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. This addresses existing shortcomings in the current requirements in that it requires that inspection frequency be part of a management strategy aimed at ensuring SG tube integrity. The proposed TS 3.4.16 Bases state that inspection frequency will be determined, in part, by operational assessments which utilize additional information on existing degradation and flaw growth rates to determine an inspection frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next SG inspection.

The NRC staff also notes, however, that any assessment or projection of the future condition of the SG tubing based on the existing condition of the tubing and anticipated flaw growth rates can involve significant uncertainty that may be difficult to conservatively and reliably bound. For this reason, proposed TS 5.5.9 supplements the performance-based requirement concerning inspection frequencies with a set of prescriptive requirements that provide added assurance that SG tube integrity will be maintained.

The proposed prescriptive requirements include a requirement that 100 percent of the tubes in each SG be inspected at the first refueling outage following SG replacement. The required scope of this inspection is substantially more restrictive than the current requirement, which requires a 3 percent sample of the total SG tube population and requires inspection of the two SGs.

For ANO-1, which will have Alloy 690 TT tubing after SG replacement in fall 2005, the proposed specification would require that 100 percent of the tubes be inspected at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months (EFPM), with the first sequential period being considered to begin at the time of the first ISI of the SGs following SG replacement. This sliding scale is intended to address the increased potential for the initiation of SCC over time. In addition, the licensee would be required to inspect 50 percent of the tubes by the refueling outage nearest the mid-point of the period and the remaining 50 percent by the refueling outage nearest the end of the period. However, no SG shall operate for more than 72 EFPM or three refueling outages (whichever is less) without being inspected.

Regardless of the type of tubing, if crack indications are found in any tube, the proposed specification requires that the next inspection for each SG for the degradation mechanism causing the crack indication shall not exceed 24 EFPM or one refueling outage (whichever is less). As a point of clarification, the proposed requirements stipulate that 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 a crack, then the indication need not be treated as such.

These proposed prescriptive requirements, in total, cannot be described simplistically as being more restrictive or less restrictive than current requirements. They are a quite different set of requirements, being generally more restrictive for SGs with low-to-moderate levels of degradation (i.e., categories C-1 to C-2 as defined in current TSs) to somewhat less restrictive for plants with extensive levels of degradation other than cracks. As previously noted, management of SCC mechanisms relative to the performance criteria poses a particular challenge compared to other degradation mechanisms. The proposed requirement to limit inspection intervals to one refueling outage to address any cracking mechanism found to be present in the SGs is a substantially more restrictive requirement than current TS requirements that apply for plants with low to moderate levels of cracked tubes and for practical purposes leads to the same inspection frequency (every refueling outage) as would be required under current TS requirements for plants with moderate to extensive levels of cracked tubes.

The proposed prescriptive requirements relating to inspection frequency have been developed based on qualitative engineering considerations and experience, reflecting the improved SCC resistance of Alloy 690 TT tubing relative to Alloy 600 TT and particularly relative to Alloy 600 MA tubing, that the potential for cracking increases with increasing time in service, and the particular challenges associated with the management of SCC with respect to satisfying the SG tube integrity performance criteria. The proposed prescriptive requirements are intended primarily to supplement the performance-based requirement that inspection frequency in conjunction with inspection scope and methods be such as to ensure SG tube integrity is maintained. This performance-based requirement must be satisfied in addition to the prescriptive requirements. The NRC staff concludes that the proposed performance-based requirement, in conjunction with the proposed prescriptive requirements, represents a significantly more effective strategy for ensuring SG tube integrity than that provided by current TS requirements and will serve to ensure that SG tube integrity is maintained between SG inspections.

3.3.4 Tube Repair Criteria

Revised TS 5.5.9 would remove the current SG-specific tube repair limits (i.e., alternate repair criteria) associated with the original SGs. The remaining tube repair limit would be retained. Specifically, the proposed specification would require that tubes found by ISI to contain flaws with a depth equal to or exceeding 40 percent of the nominal tube wall thickness be plugged. This criterion is consistent with the SG tube integrity performance criteria in that flaws not exceeding the tube repair criterion satisfy the performance criteria with allowances for flaw size measurement error and incremental crack growth between inspections. The licensee stated in the September 30, 2004, letter that analyses would be performed for the replacement SGs, prior to startup, to confirm that 40 percent is the appropriate repair limit for the SG tubes.

The TS tube repair criteria provide added assurance that SG tube integrity will be maintained, given the performance-based strategy that is also to be followed under the proposed specification. The inclusion of tube repair criteria as part of the proposed specification also ensures that the NRC staff has the opportunity to review any risk implications should the licensee propose a license amendment for alternate tube repair criteria, in conjunction with alternate SG tube integrity performance criteria, at some time in the future.

3.3.5 Monitoring of Operational Primary-to-Secondary Leakage

Proposed TS 5.5.9 would require that the SG Program include provisions for monitoring primary-to-secondary leakage. The NRC staff's evaluation of this proposal is included as part of the NRC staff's evaluation of the proposed change to TS 3.4.13, "RCS Operational LEAKAGE," in Section 3.5 of this SE.

3.4 TS 5.6.7, "Steam Generator Tube Inspection Report"

The proposed TS amendment package would revise the reporting requirements of TS 5.6.7. Currently, this specification requires that the complete results of the SG Tube Surveillance Program (i.e., the ISI results) be reported within 90 days following completion of the program and if the results of SG tube inspections fall into TS Category C-3, another report is to be submitted to the NRC prior to resumption of plant operation. The current report is required to include (1) the number and extent of the tubes inspected, (2) the location and percent of wall thickness penetration for each indication, (3) identification of tubes plugged, and (4) additional details associated with three different alternate repair criteria in-use in the original ANO-1 SGs. Under the revised requirement, a report shall be submitted within 180 days of entry into MODE 4 following a SG inspection. The report shall include:

- The scope of the inspections performed in each SG,
- active degradation mechanisms found,
- non-destructive examination techniques used for each degradation mechanism,
- location, orientation (if linear), and measured sizes (if available) of service induced indications,
- number of tubes plugged during the inspection outage for each active degradation mechanism,
- total number and percentage of tubes plugged to date, and
- the results of condition monitoring, including the results of tube pulls and in-situ testing.

This revised reporting requirement is a more comprehensive requirement than the current 90-day report and will enhance the NRC staff's ability to monitor the kinds of inspections being performed, the extent and severity of each active degradation mechanism, degradation trends (stable or getting worse), and the degree of challenge faced by the licensee in maintaining SG tube integrity. In addition, the revised reporting requirement eliminates details associated with alternate repair criteria which will no longer be applicable to the replacement SGs. The 180-day reporting requirement is adequate given that, should the SG program fail to maintain SG tube integrity as indicated by condition monitoring, this would be promptly reportable in accordance with 10 CFR 50.72 and Reference 9 allowing the NRC staff to engage in any follow-up activities that it determines to be necessary.

The specification currently requires that inspection results falling into Category C-3 shall be reported to the NRC prior to the resumption of plant operation and that the report include a description of the investigations conducted to determine the cause of the tube degradation and corrective measures taken to prevent recurrence. The proposed TS amendment package would delete both of these requirements. The NRC staff finds deletion of these requirements to be acceptable. Neither the number of tubes plugged nor the finding of Category C-3 results (i.e., 10 percent of the tubes inspected contain degradation or 1 percent of the tubes inspected satisfy the tube repair criteria) have any real bearing on whether SG tube integrity is being maintained. The NRC staff also notes that the proposed TS amendment would delete the

definition of inspection results categories in the current TSs. If the SG program is effectively maintaining SG tube integrity, tubes found to be degraded or to be pluggable will also satisfy the SG tube integrity performance criteria. The regulation at 10 CFR 50.72, in conjunction with Reference 9, require that the NRC staff be promptly notified in the event that the SG tube integrity performance criteria are not met. The NRC staff would have the opportunity under the NRC ROP to follow up on such an occurrence as warranted. The regulation at 10 CFR 50.73 requires that a Licensee Event Report be issued within 60 days of the finding that addresses, in part, the degraded condition of the tube(s) and corrective measures being taken.

Based on the foregoing, the NRC staff finds the proposed revisions to the reporting requirements to be acceptable.

3.5 TS 3.4.13, "RCS Operational LEAKAGE"

The licensee proposed several changes to the LCO, required actions, and SRs for TS 3.4.13, “RCS Operational LEAKAGE.” These changes include administrative changes to the LCO, required action statements, and SR. The proposed administrative changes included the following:

a) replacing “SG” in LCO 3.4.13.d with “steam generator (SG);”

b) deletion of existing Condition A, Required Action A, and the associated completion time;

c) relabeling Conditions B and C as A and B, respectively, due to deletion of Condition A;

d) relabeling Required Actions B and C as A and B, respectively, due to deletion of Required Action A;

e) adding “except for primary to secondary LEAKAGE” to the end of existing Condition B. Relabeled Condition A will state “RCS unidentified or identified LEAKAGE not within limits, except for primary to secondary LEAKAGE.”

f) modifying the NOTE associated with SR 3.4.13.1. “NOTE” will be changed to “NOTES,” and relabeled as “1” and a second note, labeled as “2”, will be added, which will state “Not applicable to primary to secondary LEAKAGE.”

The NRC staff has reviewed these administrative changes and finds them acceptable. In particular, the addition of “except for primary to secondary LEAKAGE” to existing Condition B and SR 3.4.13.1 Note 2 are considered to be administrative changes because these changes support the more restrictive addition of primary-to-secondary LEAKAGE to existing Condition C and SR 3.4.13.2. The need for Note 2 with respect to SR 3.4.13.1 (i.e., not applicable to primary-to-secondary LEAKAGE) and for the proposed new SR 3.4.13.2, which deals with primary-to-secondary LEAKAGE, is discussed in the proposed revision to the Bases for TS 3.4.13.2. The revised Bases state that SR 3.4.13.1 is not applicable to primary-to-secondary leakage because leakage rates of 150 gpd or less cannot be accurately measured by an RCS water inventory balance.

3.5.1 New TS 3.4.13 Condition B Primary-to-Secondary LEAKAGE

The primary-to-secondary leakage limit provides assurance against tube rupture at normal operating and faulted conditions. This, together with the allowable accident induced leakage limit, helps to ensure that the dose contribution from tube leakage will be limited to less than the 10 CFR Part 100 dose guidelines and GDC 19 dose limits or other NRC-approved licensing basis for postulated faulted events. As previously stated, existing Conditions B and C were relabeled as Conditions A and B, since the original Condition A was eliminated. The licensee proposed to add an additional OR statement to new Condition B with regards to primary-to-secondary LEAKAGE. Also, the licensee proposed to eliminate the reference to Condition B since it was relabeled as Condition A. As proposed, new Condition B would state:

> "Required Action and associated Completion Time of Condition A not met.
>
> OR
>
> Pressure boundary LEAKAGE exists.
>
> OR
>
> Primary to secondary LEAKAGE not within limit."

The current requirements for new Condition A have a completion time of 18 hours to reduce LEAKAGE (other than pressure boundary LEAKAGE) to within limits after which new Condition B (plant shutdown) must be entered. The TS limit is more restrictive than the current requirements in that if primary-to-secondary leakage exceeds 150 gpd, then a plant shutdown must be commenced without an allowance to reduce leakage, as provided in original Condition A. The revised Condition B would require the reactor to be in MODE 3 in 6 hours and MODE 5 in 36 hours if primary-to-secondary leakage is not within limits. As discussed in Section 3.5 above, the licensee has excluded primary-to-secondary leakage from new Condition A. The NRC staff has reviewed the proposed change to new Condition B. These changes are additional restrictions on plant operations that enhance safety; therefore, the NRC staff has concluded that the addition of the primary-to-secondary leakage OR statement to new Condition B is acceptable.

3.5.2 SRs - Primary-To-Secondary Leakage

SR 3.4.13.1 currently requires verification that RCS operational LEAKAGE is within limits by performance of RCS water inventory balance. The accompanying Basis states that primary-to-secondary leakage is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems. These Bases further state that the RCS water inventory balance must be met with the reactor at steady state operating conditions and near operating pressure. As previously discussed in Section 3.5 of this SE, the licensee has proposed adding a note to this SR stating that this particular SR is not applicable to primary-to-secondary LEAKAGE. The licensee would revise the accompanying Bases justifying this change, namely, LEAKAGE of 150 gpd cannot be measured accurately by an RCS water inventory balance. The licensee has proposed a new SR, SR 3.4.13.2, which would verify with a frequency of 72 hours that primary-to-secondary leakage does not exceed the 150 gpd LCO limit. The NRC staff believes this to be acceptable

and in accordance with 10 CFR 50.36(c)(3). The revised requirement would not specify the specific method to be employed; however, it would require that the SG Program include provisions for monitoring primary-to-secondary leakage. There is a variety of methods that can be used, and the NRC staff concludes there is no need to tie this surveillance to a specific method in order to ensure that the plant is operated safely and within its LCO limits. The licensee would state in the accompanying Bases that the primary-to-secondary leakage measurement uses continuous process radiation monitors or radio chemical grab sampling. The NRC staff notes that the EPRI PWR Primary-to-Secondary Leak Guidelines provide extensive guidance to this effect.

The accompanying Bases would also state that primary-to-secondary LEAKAGE is measured against the 150 gpd limit under room temperature conditions as described in the EPRI PWR Primary-to-Secondary Leak Guidelines. The Bases state that primary-to-secondary LEAKAGE is a factor in the radioactivity releases resulting from a steam line break (SLB) accident. The ANO-1 safety analysis for an SLB accident assumes 1 gpm of primary-to-secondary LEAKAGE in one generator as an initial condition. The NRC staff concludes that measurement of operational primary-to-secondary LEAKAGE under room temperature conditions relative to the 150 gpd operational limit is acceptable since it ensures that LEAKAGE under hot normal operating conditions will be less than assumed in the ANO-1 safety analysis and, thus, is consistent with 10 CFR 50.36(c)(ii). However, satisfying the 150 gpd operational limit is not sufficient for ensuring that leakage under accident conditions will remain within the values assumed in the licensing basis accident analyses. This must be demonstrated by periodic condition monitoring as discussed in Section 3.3.1.2 and 3.3.2 of this safety evaluation.

The new SR, SR 3.4.13.2, with respect to primary-to-secondary leakage, replaces the current SR 3.4.13.2, which involved verifying SG tube integrity in accordance with the SG Tube Surveillance Program. As discussed earlier in this SE, TS 5.5.9, “Steam Generator Tube Surveillance Program,” would be replaced by TS 5.5.9, “Steam Generator Program.” The SR to verify SG tube integrity would be addressed in the proposed new TS 3.4.16, “Steam Generator (SG) Tube Integrity,” SRs.

Based on the above, the NRC staff concludes that the proposed revisions to SR 3.4.13.1 and SR 3.4.13.2 are acceptable. Although the TS Bases are controlled under the auspices of 10 CFR 50.59 and TS 5.5.14, TS Bases Control Program, the NRC staff finds the proposed changes to the proposed TS 3.4.13 Bases to be acceptable.

3.6 TS 1.1, “Definition”

Lastly, the proposed changes to TS 1.1, “Definition,” are editorial changes to the definition of LEAKAGE. The existing ANO-1 TS 1.1 contains (in part) definitions for “Identified LEAKAGE” and “Pressure Boundary LEAKAGE.” The ANO-1 definitions refer to SG LEAKAGE but that term is not used in the TS or Bases. The term used in the TS and Bases is “primary to secondary LEAKAGE.” Therefore, the licensee proposed modifying the two definitions currently in the TS to reflect primary-to-secondary leakage.

Based on the editorial nature of the change, the NRC staff concludes that the proposed revisions to TS 1.1 are acceptable.

3.7 Impact on Previously Analyzed Radiological Consequences of DBAs

The license amendment request makes changes to the TSs regarding the ANO-1 SG tube ISI program. The licensee's proposed changes to the ANO-1 TSs will be applicable to replacement SGs, which are currently planned to be installed in refueling outage 1R19, scheduled for the fall of 2005. The licensee provided background information on the replacement SGs and stated that the replacement SGs are similar in thermal and hydraulic performance to the original SGs. The licensee also determined pursuant to 10 CFR 50.59 that the SGs could be replaced without needing prior NRC approval. As indicated in the September 30, 2004, application, the physical dimensions, secondary coolant volume, and thermal and hydraulic performance for the replacement SGs are similar to those of the original SGs, for which the current DBA dose analyses have been performed. In other words, the DBA analysis assumptions for SG secondary coolant steam release and primary-to-secondary leakage for the replacement SGs would be bounded by the current assumptions for the original SGs. Therefore, the license amendment request did not provide DBA dose analyses for operation at ANO-1 with the replacement SGs, and the NRC staff did not review this area.

The NRC staff finds that the proposed changes to the ANO-1 TSs for the SG tube ISI program do not affect the inputs, assumptions or methodology used in the current DBA dose analyses, and the licensee did not make any revisions to the ANO-1 current licensing basis radiological dose analyses. In addition, the NRC staff does not find that any of the proposed TS changes contradict the current ANO-1 licensing basis with respect to the analysis of radiological consequences of DBAs.

3.8 Technical Evaluation Summary

The proposed TS amendment package establishes a programmatic, largely performance-based regulatory framework for ensuring SG tube integrity is maintained. The NRC staff finds that it addresses key shortcomings of the current framework by ensuring that SG programs are focused on accomplishing the overall objective of maintaining SG tube integrity. It incorporates performance criteria for evaluating SG tube integrity that the NRC staff finds are consistent with the structural margins and the degree of leak tightness assumed in the current plant licensing basis. The NRC staff finds that maintaining these performance criteria provides reasonable assurance that the SGs can be operated safely without increase in risk.

The revised TSs contain limited details concerning how the SG Program is to achieve the required objective of maintaining SG tube integrity, the intent being that the licensee will have the flexibility to determine the specific strategy for meeting this objective. However, the NRC staff finds that the revised TSs include sufficient regulatory constraints on the establishment and implementation of the SG Program to provide reasonable assurance that SG tube integrity will be maintained.

Failure to meet the performance criteria will be reportable pursuant to 10 CFR 50.72 and 50.73. The NRC ROP provides a process by which the NRC staff can verify that the licensee has identified any SG Program deficiencies that may have contributed to such an occurrence and that appropriate corrective actions have been implemented.

Regarding the impact of the proposed changes to the SG tube ISI program at ANO-1 on the previously analyzed radiological consequences of DBAs, the NRC staff finds with reasonable

assurance that the current licensing basis radiological dose consequences analyses for DBAs remain bounding for the proposed changes to the TSs. Therefore, the proposed TS changes are acceptable with regard to the radiological consequences of postulated DBAs.

In summary, the NRC staff finds that the ANO-1 TS amendment request conforms to the requirements of 10 CFR 50.36 and establishes a TS framework that will provide reasonable assurance that SG tube integrity is maintained without undue risk to public health and safety.

4.0 REGULATORY COMMITMENTS

The licensee's September 30, 2004, application contains the following regulatory commitments:

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE (If Required)
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
Entergy will revise the ANO-1 SAR to reference NEI 97-06 and the Steam Generator Program TSs.	X		Prior to next scheduled SAR submittal after operating license Amendment approval
The revised TS requirements under TSTF 449, Draft Revision 2 require those loads that significantly affect burst or collapse 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. These loads, as well as the other analyses to support a 40% plugging limit, will be analyzed for the RSGs under the existing ANO-1 licensing basis. These analyses will be performed and documented under the requirements of 10 CFR 50.59.	X		Prior to startup from 1R19

The NRC staff finds that reasonable controls for the implementation and for subsequent evaluation of proposed changes pertaining to the above regulatory commitments are best provided by the licensee’s administrative processes, including its commitment management program. The above regulatory commitments do not warrant the creation of a regulatory requirement (item requiring prior NRC approval of subsequent changes).

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Arkansas State official was notified of the proposed issuance of the amendment. The State official had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (69 FR 64987 dated November 9, 2004, and 70 FR 29790 dated May 24, 2005). The amendment also relates to changes in recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES

7) Letter, R.E. Beedle, NEI, to L. J. Callan, NRC, December 16, 1997, transmitting NEI 97-06 (Original), “Steam Generator Program Guidelines.” ADAMS Accession No. 9801050189.

8) NEI 97-06, Revision 1, “Steam Generator Program Guidelines,” January 2001. ADAMS Accession No. ML010430054.

9) SECY-00-0078, “Status and Plans for Revising the Steam Generator Tube Integrity Regulatory Framework,” March 30, 2000. ADAMS Accession No. ML003691745.

10) Letter, Sean Peters, NRC, to L. M. Stinson, Vice President, Southern Nuclear Operating Company, “Joseph M. Farley Nuclear Plant, Units 1 and 2, re: Issuance of Amendments to Facilitate Implementation of Industry Initiative NEI 97-06, Steam Generator Program Guidelines,” dated September 10, 2004. ADAMS Accession No. ML042570427.

11) Draft Regulatory Guide 1.121, “Bases for Plugging Degraded PWR Steam Generator Tubes,” August 1976. ADAMS Accession No. 8808230027.

12) Memorandum dated September 8, 1999, to W. H. Bateman, Chief, EMCB, NRR, NRC from J. W. Andersen, EMCB, NRR, NRC, “Summary of August 27, 1999, Senior Management Meeting with NEI/EPRI/Industry to Discuss Issues Involving Implementation of NEI 97-06.” This memorandum encloses the industry white paper entitled, “Deterministic Structural Performance Criterion Pressure Loading Definition.” ADAMS Accession No. 9909170231.

13) Memorandum dated May 19, 2004, from J. L. Birmingham, Project Manager, NRR, NRC to Cathy Haney, Program Director, Policy and Rulemaking Program, Division of Regulatory Improvement Programs, NRR, NRC, “Summary of May 14, 2004 Meeting with Nuclear Energy Institute (NEI) on Status of Steam Generator Structural Integrity Performance Criteria.” ADAMS Accession No. ML041540500.

14) NUREG-1570, “Risk Assessment of Severe Accident -Induced Steam Generator Tube Rupture,” March 1998. ADAMS Accession No. 9803310390.

15) NUREG-1022, Rev 2, “Event Reporting Guidelines 10 CFR 50.72 and 50.73,” October 31, 2000[1]. ADAMS Accession No. ML003762595.

10) NUREG-1649, Rev 3, “Reactor Oversight Process,” July 2000. ADAMS Accession No. ML003738931.

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(1)On February 18, 2004, a Federal Register notice (69 FR 7661) was issued requesting comments on the NRC's intent to issue an errata to Revision 2 of NUREG-1022, "Event Reporting Guidelines 10 CFR 50.72 and 50.73." The errata would indicate that SG tube degradation is considered serious if either of the two criteria specified in Section 3.2.4(A)(3) of NUREG-1022 (i.e., the structural and accident leakage performance criteria), Revision 2, are not satisfied.

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