WOLF CREEK
NUCLEAR OPERATING CORPORATION

Britt T. McKinney
Site Vice President

OCT 17 2003

WO 03-0051

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

Subject: Docket No. 50-482: Revision to Technical Specification 5.5.6, "Containment Tendon Surveillance Program," and Technical Specification 5.5.16, "Containment Leakage Rate Testing Program"

Gentlemen:

Wolf Creek Nuclear Operating Corporation (WCNOC) hereby transmits an application for amendment to Facility Operating License No. NPF-42 for the Wolf Creek Generating Station (WCGS) in accordance with the provisions of 10 CFR 50.90. The proposed amendment would revise Technical Specification (TS) Section 5.5.6, "Containment Tendon Surveillance Program," and Section 5.5.16, "Containment Leakage Rate Testing Program," for consistency with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class CC. This regulation requires licensees to update their containment inservice inspection requirements in accordance with Subsections IWE and IWL of Section XI, Division I of the ASME Boiler and Pressure Vessel Code as limited by 10 CFR 50.55a(b)(2)(vi) and modified by 10 CFR 50.55a(b)(2)(viii) and 10 CFR50.55a(b)(2)(ix).

Attachments I through IV provide the Evaluation, Markup of Technical Specifications, Retyped Technical Specifications, and Proposed TS Bases Changes, respectively, in support of this amendment request. Attachment IV contains the TS Bases changes (for information only) to assist the staff in its review of the proposed changes. Revision to the TS Bases will be implemented pursuant to the TS Bases Control Program, TS 5.5.14, upon implementation of this license amendment. Attachment V contains a list of commitments.

This amendment application was reviewed by the Plant Safety Review Committee and the Nuclear Safety Review Committee. In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated Kansas State official.

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WCNOC is submitting this license amendment request in conjunction with an industry consortium of six stations as a result of a mutual agreement known as Strategic Teaming and Resource Sharing (STARS). The STARS group consists of the six stations operated by TXU Generation Company LP, Union Electric Company, WCNOC, Pacific Gas and Electric Company, STP Nuclear Operating Company, and Arizona Public Service Company. Other members of the group are expected to submit license amendment requests similar to this one. Due to differences between the STARS plants, there may be some differences in the plant license amendment requests, particularly for the information provided in Attachment I.

WCNOC requests approval of the proposed amendment by April, 2004 to support performance of the containment tendon surveillances during Summer, 2004. It is anticipated that the license amendment, as approved, will be effective upon issuance, to be implemented within 90 days from the date of issuance. Please contact me at (620) 364-4112 or Mr. Kevin Moles at (620) 364-4126 for any questions you may have regarding this application.

Sincerely,

Britt T. McKinney

Britt T. McKinney

BTM/rlr

Attachments:

I	-	Evaluation
II	-	Markup of Technical Specification pages
III	-	Retyped Technical Specification pages
IV	-	Proposed TS Bases Changes (for information only)
V		List of Commitments

cc: V. L. Cooper (KDHE), w/a
J. N. Donohew (NRC), w/a
D. N. Graves (NRC), w/a
B. S. Mallett (NRC), w/a
Senior Resident Inspector (NRC), w/a

STATE OF KANSAS)
) SS
COUNTY OF COFFEY)

Britt T. McKinney, of lawful age, being first duly sworn upon oath says that he is Site Vice President of Wolf Creek Nuclear Operating Corporation; that he has read the foregoing document and knows the contents thereof; that he has executed the same for and on behalf of said Corporation with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By Britt T. McKinney

Britt T. McKinney
Site Vice President

SUBSCRIBED and sworn to before me this 17 day of October 2003.

Rhonda L. Gleue

Notary Public

NOTARY PUBLIC STATE OF KANSAS RHONDA L. GLEUE My Appt. Exp.

Expiration Date May 11, 2006

EVALUATION

1.0 DESCRIPTION

The proposed amendment revises Technical Specification (TS) Section 5.5.6, "Containment Tendon Surveillance Program," and Section 5.5.16, "Containment Leakage Rate Testing Program," for consistency with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class CC. This regulation requires licensees to update their containment inservice inspection requirements in accordance with Subsections IWE and IWL of Section XI, Division I of the ASME Boiler and Pressure Vessel Code as limited by 10 CFR 50.55a(b)(2)(vi) and modified by 10 CFR 50.55a(b)(2)(viii) and 10 CFR50.55a(b)(2)(ix).

2.0 PROPOSED CHANGE

The proposed change will revise:

- Technical Specification 5.5.6

This specification is revised to indicate that the Containment Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, except where an exemption or relief has been authorized by the NRC. Additionally, the provisions of Surveillance Requirement (SR) 3.0.2 are deleted from this specification.

- Technical Specification 5.5.16

This specification is revised to add the following exceptions to Regulatory Guide 1.163, "Performance- Based Containment Leak-Testing Program,"

"1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC."

The TS Bases for SR 3.6.1.1 and SR 3.6.1.2 are also revised for consistency with the requirements of the ASME Code Section XI, Subsection IWL and applicable addenda as required by 10 CFR 50.55a.

3.0 BACKGROUND

On January 7, 1994, the Nuclear Regulatory Commission (NRC) published a proposed amendment to the regulations to incorporate by reference the 1992 Edition with the 1992 Addenda of Subsections IWE and IWL of Section XI, Division I of the ASME Boiler and Pressure Vessel Code (the Code). The final rule, Subpart 50.55a(g)(6)(ii)(B) of Title 10 of the Code of Federal Regulations (10 CFR), became effective on September 9, 1996, and requires licensees to implement Subsections IWE and IWL, with specified modifications and limitations, by September 9, 2001.

The containment consists of a prestressed, reinforced concrete, cylindrical structure with a hemispherical dome. The Post-tensioning System used for the shell and dome of the containment employs tendons. Each tendon consists of high strength steel wires and anchoring components. The prestressing load is transferred, by cold formed button heads on the ends of the individual wires through stressing washers, to steel bearing plates embedded in the structure. The unbonded tendons are installed in tendon ducts and tensioned in a predetermined sequence.

4.0 TECHNICAL ANALYSIS

Technical Specification 5.5.6, "Containment Tendon Surveillance Program," states in part, "The Containment Tendon Surveillance Program, and its inspection frequencies and acceptance criteria shall be in accordance with Wolf Creek Generation Station position on draft Revision 3 of Regulatory Guide 1.35, dated April, 1979." As identified above, 10 CFR 50.55a(g)(4) requires licensees to update their containment inservice inspection requirements in accordance with Subsections IWE and IWL of Section XI, Division I of the ASME Boiler and Pressure Vessel Code as limited by 10 CFR 50.55a(b)(2)(vi) and modified by 10 CFR 50.55a(b)(2)(viii) and 10 CFR50.55a(b)(2)(ix). The requirements in 10 CFR 50.55a(g)(4) and ASME Code Section XI, Subsection IWL do not reference Regulatory Guide 1.35, Revision 3. As such, the TS are inconsistent with the requirements of 10 CFR 50.55a.

10 CFR 50.55a(g)(5)(ii) states, in part: "If a revised inservice inspection program for a facility conflicts with the technical specification for the facility, the licensee shall apply to the Commission for Amendment of the technical specifications to conform the technical specification to the revised program." Based on the requirements in 10 CFR 50.55a, WCNOC is required to update the technical specifications. The containment inservice inspection programs are required to be in accordance with ASME Code Section XI, Subsection IWL as modified by 10 CFR 50.55a(b)(2)(viii), except where an exemption or relief has been authorized by the NRC.

Additionally, since the tendon inspection frequencies will be in accordance with ASME Section XI, Subsection IWL, the provisions of SR 3.0.2 are no longer applicable and are deleted from Technical Specification 5.5.6. As discussed in the Technical Specification Bases for SR 3.0.2, the requirements of regulations take precedence over the Technical Specifications. As such, 10 CFR 50.55a requires the implementation of ASME Section XI, Subsection IWL and specifies the requirements for extending inspection frequencies.

Technical Specification 5.5.16 contains requirements for the Containment Leakage Rate Testing Program, and it specifies that the program shall be in accordance with the guidelines contained in Regulatory Guide 1.163. Regulatory Position C.3 of the regulatory guide states that "Section 9.2.1, "Pretest Inspection and Test Methodology," of NEI 94-01 provides guidance

for the visual examination of accessible interior and exterior surfaces of the containment system for structural problems. These examinations should be conducted prior to initiating a Type A test, and during two other refueling outages before the next Type A test if the interval for the Type A test has been extended to 10 years, in order to allow for early uncovering of evidence of structural deterioration." There are no specific requirements in NEI 94-01 for the visual examination except that it is to be a general visual examination of accessible interior and exterior surfaces of the primary containment components.

In addition to the requirements of Regulatory Guide 1.163 and NEI 94-01, the concrete surfaces of the containment must be visually examined in accordance with the ASME Section XI Code, Subsection IWL, and the liner plate inside containment must be visually examined in accordance with Subsection IWE. The frequency of visual examination of the concrete surfaces per Subsection IWL is once every five years, and the frequency of visual examination of the liner plate per Subsection IWE is, in general, three visual examinations over a 10-year period. The visual examinations performed pursuant to Subsection IWL may be performed at any time during power operation or during shutdown, and the visual examinations performed pursuant to Subsection IWE are performed during refueling outages since this in the only time that the liner plate is fully accessible.

In addition, the visual examinations performed pursuant to Subsections IWL and IWE are more rigorous than those performed pursuant to Regulatory Guide 1.163 and NEI 94-01. For example, Subarticle IWE-2320 requires the general visual examination to be the responsibility of an individual who is knowledgeable in the requirements for design, inservice inspection, and testing of Class MC and metallic liners of Class CC components. Subsection IWE, Subarticle-2330 requires the examination to be performed either directly or remotely, by an examiner with visual acuity sufficient to detect evidence of degradation.

Similarly, Subarticle IWL-2320 states that:

> "The Responsible Engineer shall be a Registered Professional Engineer experienced in evaluating the condition of structural concrete. The Responsible Engineer shall have knowledge of the design and Construction Codes and other criteria used in design and construction of concrete containments in nuclear power plants.
>
> The Responsible Engineer shall be responsible for the following:
>
> (a) development of plans and procedures for examination of concrete surfaces;
> (b) approval, instruction, and training of concrete examination personnel;
> (c) evaluation of examination results;
> (d) preparation or review of Repair/Replacement Plans and procedures;
> (e) review of procedures for pressure tests following repair/replacement activities;
> (f) submittal of report to the Owner documenting results of examinations, repair/replacement activities, and pressure tests."

Based on the above, the Responsible Engineer will ensure that a comprehensive visual examination of the concrete is performed in accordance with Code requirements except where relief has been granted by the NRC. Furthermore, with respect to examinations performed pursuant to both Subsections IWL and IWE, visual examinations of both the concrete surfaces and the liner plate must be reviewed by an Inspector employed by a State or municipality of the United States or an Inspector regularly employed by an insurance company authorized to write boiler and pressure vessel insurance, in accordance with IWA-2110 and IWA-2120. The combination of the Code requirements for the rigor of the visual examinations plus the third-

party review will more than offset the fact one fewer visual examination of the concrete will be performed during a 10-year interval. The fact that the concrete visual examination pursuant to Subsection IWL may be performed during power operation as opposed to during a refueling outage will have no effect on the quality of the examination and will provide flexibility in scheduling of the visual examinations.

5.0 REGULATORY ANALYSIS

This section addresses the standards of 10 CFR 50.92 as well as the applicable regulatory requirements and acceptance criteria.

5.1 NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC)

The proposed amendment revises Technical Specification (TS) Section 5.5.6, "Containment Tendon Surveillance Program," and Section 5.5.16, "Containment Leakage Rate Testing Program," for consistency with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class CC. The proposed change does not involve a significant hazards consideration for Wolf Creek Generating Station based on the three standards set forth in 10 CFR 50.92(c) as discussed below:

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

Response: No

The proposed change revises the TS administrative controls programs for consistency with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class CC. The revised requirements do not affect the function of the containment post-tensioning system components. The post-tensioning systems are passive components whose failure modes could not act as accident initiators or precursors.

The proposed change affects the frequency of visual examinations that will be performed for the concrete surfaces of the containment for the purpose of the Containment Leakage Rate Testing Program. In addition, the proposed change allows those examinations to be performed during power operation as opposed to during a refueling outage. The frequency of visual examinations of the concrete surfaces of the containment and the mode of operation during which those examinations are performed has no relationship to or adverse impact on the probability of any of the initiating events assumed in the accident analyses. The proposed change would allow visual examinations that are performed pursuant to NRC approved ASME Section XI Code requirements (except where relief has been granted by the NRC) to meet the intent of visual examinations required by Regulatory Guide 1.163, without requiring additional visual examinations pursuant to the Regulatory Guide. The intent of early detection of deterioration will continue to be met by the more rigorous requirements of the Code required visual examinations. As such, the safety function of the containment as a fission product barrier is maintained.

The proposed change does not impact any accident initiators or analyzed events or assumed mitigation of accident or transient events. They do not involve the addition or removal of any equipment, or any design changes to the facility.

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

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

Response: No

The proposed change revises the TS administrative controls programs for consistency with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class CC. The function of the containment post-tensioning system components are not altered by this change. The change affects the frequency of visual examinations that will be performed for the concrete surfaces containments. In addition, the proposed change allows those examinations to be performed during power operation as opposed to during a refueling outage. The proposed change does not involve a modification to the physical configuration of the plant (i.e., no new equipment will be installed) or change in the methods governing normal plant operation. The proposed change will not impose any new or different requirements or introduce a new accident initiator, accident precursor, or malfunction mechanism. Additionally, there is no change in the types or increases in the amounts of any effluent that may be released off-site and there is no increase in individual or cumulative occupational exposure.

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

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

Response: No

The proposed change revises the TS administrative controls programs for consistency with the requirements of 10 CFR 50.55a(g)(4) for components classified as Code Class CC. The function of the containment post-tensioning system components are not altered by this change. The change affects the frequency of visual examinations that will be performed for the concrete surfaces containments. In addition, the proposed change allows those examinations to be performed during power operation as opposed to during a refueling outage. The safety function of the containment as a fission product barrier will be maintained.

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

Conclusion:

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

5.2 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

The regulatory basis for TS 3.6.1, "Containment," is to ensure that the primary containment is capable of remaining leak-tight following a loss of coolant accident. This ensures that offsite radiation exposures are maintained within the limits of 10CFR100.

10 CFR 50, Appendix A, General Design Criterion 16, "Design," requires that reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as the postulated accident conditions require.

This TS change will not reduce the leak-tightness of the containment. Therefore, based on the considerations discussed above:

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) Issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

WCNOC has determined that the proposed amendment would change requirements with respect to the installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, WCNOC has determined that the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amount of effluent that may be released offsite, or (iii) a significant increase in the individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22 (c)(9). Therefore, pursuant to 10 CFR 51.22 (b), an environmental assessment of the proposed amendment is not required.

7.0 REFERENCES

1. 10 CFR 50.55a.

2. Regulatory Guide 1.163, "Performance-Based Containment Leak-Testing Program."

3. Letter dated January 18, 2000, to W. R. McCollum, Jr., Duke Energy Corporation, "Oconee Nuclear Station Units 1, 2, and 3 RE: Issuance of Amendments (TAC Nos. MA6568, MA6569, and MA6570)." Amendment Nos. 310.

4. Letter dated June 6, 2001, to J. B. Beasley, Jr., Southern Nuclear Operating Company, Inc, "Vogtle Electric Generating Plant, Units 1 and 2 RE: Issuance of Amendments (TAC Nos. MB1097 and MB1098)." Amendment Nos. 122 and 100.

5. Letter dated January 30, 2001, to C. H. Cruse, Constellation Nuclear, "Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 RE: Containment Tendon Surveillance Program – Amendment (TAC Nos. MB0011 and MB0012)." Amendment Nos. 240 and 214.

The NRC issued License Amendments 310, 310, and 310 for Oconee Nuclear Station, Units 1, 2 and 3 on January 18, 2000, License Amendments 240 and 214 for Calvert Cliffs Nuclear Power Plant, Units 1 and 2 on January 30, 2001, and License Amendments 122 and 100 for Vogtle Electric Generating Plant, Units 1 and 2 on June 6, 2001. The Amendments for Oconee and Calvert Cliffs changed the Concrete Containment Tendon Surveillance Program from being in accordance with Regulatory Guide 1.35 to ASME Code Section XI, Subsection IWL. Additionally, the provisions of SR 3.0.2 were deleted. The Amendments for Vogtle added an exception to the Containment Leakage Rate Testing Program for Regulatory Guide 1.163. The exception allows the use of ASME Code Section XI, Subsection IWL for the inspection of concrete surfaces.

ATTACHMENT II

MARKUP OF TECHNICAL SPECIFICATION PAGES

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

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

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

k. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the USAR, Section 3.9(N), cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Containment Tendon Surveillance Program

This program provides controls for monitoring tendon performance, including the effectiveness of the tendon corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial plant operation as well as periodic testing thereafter. The Containment Tendon Surveillance Program, and its inspection frequencies and acceptance criteria, shall be in accordance with ~~Wolf Creek Generating Station position on draft Revision 3 of Regulatory Guide 1.35, dated April, 1979.~~

INSERT A

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

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975. In lieu of Position C.4.b(1) and C.4.b(2), conduct a qualified in place UT examination over the volume from the inner bore

(continued)

INSERT A

Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, except where an exemption or relief has been authorized by the NRC.

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995. INSERT B

b. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 48 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.20% of containment air weight per day.

d. Leakage rate acceptance criteria are:

1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests;

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

b) For each door, leakage rate is $\leq 0.005\ L_a$ when pressurized to $\geq$ 10 psig.

e. The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

f. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

(continued)

INSERT B

as modified by the following exceptions:

1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

ATTACHMENT III

RETYPED TECHNICAL SPECIFICATION PAGES

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

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

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

k. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the USAR, Section 3.9(N), cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Containment Tendon Surveillance Program

This program provides controls for monitoring tendon performance, including the effectiveness of the tendon corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial plant operation as well as periodic testing thereafter. The Containment Tendon Surveillance Program, and its inspection frequencies and acceptance criteria, shall be in accordance with Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a, except where an exemption or relief has been authorized by the NRC.

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

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975. In lieu of Position C.4.b(1) and C.4.b(2), conduct a qualified in place UT examination over the volume from the inner bore

(continued)

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 Containment Leakage Rate Testing Program

a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exceptions:

   1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

   2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.

b. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 48 psig.

c. The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.20% of containment air weight per day.

d. Leakage rate acceptance criteria are:

   1. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests;

(continued)

5.5 Programs and Manuals

5.5.16 Containment Leakage Rate Testing Program (continued)

2. Air lock testing acceptance criteria are:

a) Overall air lock leakage rate is ≤ 0.05 L_a when tested at ≥ P_a.

b) For each door, leakage rate is ≤ 0.005 L_a when pressurized to ≥ 10 psig.

e. The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

f. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.5.17 Reactor Vessel Head Closure Bolt Integrity

This program provides the requirements to support normal plant operation with one reactor vessel head closure bolt less than fully tensioned for one operating cycle. The provisions of this program shall be implemented when a head closure bolt becomes stuck in a partially inserted position such that the amount of thread engagement is not sufficient to take the tensioning loads without damage to the vessel threads or a bolt is not capable of being inserted into the bolt hole.

Prior to operation with one reactor vessel head closure bolt less than fully tensioned, the following conditions shall apply:

a. The circumstances associated with the less than fully tensioned closure bolt will be verified to be bounded by the analysis that was referenced in the letter dated September 15, 2000 (WO 00-0036).

b. A review of the results of the visual examinations performed on the closure bolts shall be performed to ensure that there is no indication of sufficient degradation of closure bolts that could affect the conclusions of Specification 5.5.17a. above.

Within 30 days following startup of the plant, a report shall be submitted to the Commission identifying the circumstances for operation with one reactor vessel head closure bolt less than fully tensioned.

Operation with the same reactor vessel head closure bolt less than fully tensioned shall be limited to one operating cycle (i.e., until the next refueling outage).

ATTACHMENT IV

PROPOSED TS BASES CHANGES (for information only)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.1.1

INSERT B-1

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of the Containment Leakage Rate Testing Program. Failure to meet air lock and purge valve with resilient seal leakage limits specified in LCO 3.6.2 and LCO 3.6.3 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test is required to be < 0.6 L_a for combined Type B and C leakage, and ≤ 0.75 L_a for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 L_a. At ≤ 1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis.

SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

SR 3.6.1.2

This SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency ~~are consistent with the recommendations of Regulatory Guide 1.35 (Ref. 4).~~ in accordance with ASME Code Section XI, Subsection IWL (Ref. 4), and applicable addenda as required by 10 CFR 50.55a.

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. USAR, Chapter 15.
3. USAR, Section 6.2.
4. ~~Regulatory Guide 1.35, Draft Revision 3, April 1979.~~ ASME Code Section XI, Subsection IWL.

INSERT B-1

The containment concrete visual examinations may be performed during either power operation, e.g., performed concurrently with other containment inspection-related activities such as tendon testing, or during a maintenance/refueling outage. The visual examinations of the steel liner plate inside containment are performed during maintenance or refueling outages since this is the only time the liner plate is fully accessible.

LIST OF COMMITMENTS

The following table identifies those actions committed to by Wolf Creek Nuclear Operating Corporation in Attachment I to this letter. Other statements in Attachment I to this letter are not considered to be regulatory commitments. Please direct questions regarding these commitments to Mr. Kevin Moles, Manager Regulatory Affairs at Wolf Creek Generating Station, (620) 364-4126.

COMMITMENT	Due Date/Event
Revision to the TS Bases will be implemented pursuant to the TS Bases Control Program, TS 5.5.14, upon implementation of this license amendment.	Upon implementation of amendment
The license amendment, as approved, will be implemented within 90 days from the date of issuance.	Within 90 days of date of issuance of amendment