Omaha Public Power District

444 South 16th Street Mall
Omaha NE 68102-2247

October 8, 2002
LIC-02-0103

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

References: 1. Docket No. 50-285
2. Improved Standard Technical Specification (ITS) for Combustion Engineering Plants, NUREG-1432, Revision 2

SUBJECT: Fort Calhoun Station Unit No. 1 License Amendment Request, Relocation of Prestressed Containment Tendon Surveillances to the USAR

Pursuant to 10 CFR 50.90, Omaha Public Power District (OPPD) hereby transmits an application for amendment to the Fort Calhoun Station Unit 1 (FCS) Operating License. Attachment 1 provides the No Significant Hazards Evaluation and the technical bases for this requested change to the Technical Specifications (TS). Attachments 2 and 3 contain marked-up and clean-typed Technical Specification pages reflecting the requested Technical Specification and Basis changes.

The proposed amendment relocates the requirements of TS 3.5(5) for testing Prestressed Concrete Containment Tendons to the FCS Updated Safety Analysis Report (USAR). This proposed amendment also adds a TS requirement for a Containment Tendon Testing Program as TS 5.21 consistent with that presented in Section 5.5 of Reference 2.

OPPD requests approval of the proposed amendment by March 15, 2003. OPPD requests 120 days to implement this amendment. No commitments are made to the NRC in this letter.

I declare under penalty of perjury that the foregoing is true and correct. (Executed on October 8, 2002)

A001

If you have any questions or require additional information, please contact Dr. R. L. Jaworski at (402) 533-6833.

Sincerely,

D. J. Bannister
Manager Fort Calhoun Station

DJB/TRB/trb

Attachments:
1. Fort Calhoun Station's Evaluation
2. Markup of Technical Specification Pages
3. Clean-Typed Technical Specification Pages

c: E. W. Merschoff, NRC Regional Administrator, Region IV
A. B. Wang, NRC Project Manager
J. G. Kramer, NRC Senior Resident Inspector
Division Administrator - Public Health Assurance, State of Nebraska
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Attachment 1

Fort Calhoun Station's Evaluation
For
Relocation of Prestressed Containment Tendon Surveillances
to the USAR

1.0 INTRODUCTION

This letter is a request to amend Operating License DPR-40 for the Fort Calhoun Station (FCS) Unit No. 1.

Omaha Public Power District (OPPD) proposes to relocate the requirements of TS 3.5(5) for testing Prestressed Concrete Containment Tendons to the FCS Updated Safety Analysis Report (USAR). This proposed amendment also adds a TS requirement for a Containment Tendon Testing Program as TS 5.21 consistent with that presented in Section 5.5 of Reference 10.1. This is acceptable since testing of containment tendons in accordance with ASME Boiler and Pressure Vessel Code, Section IX, Subsections IWE and IWL is specified in 10 CFR 50.55a. This change eliminates duplication of federal regulations. Therefore, this system does not meet the criteria set forth in 10 CFR 50.36(c)(2)(ii) for inclusion in the TS, and the requirements will be relocated to the USAR.

2.0 DESCRIPTION OF PROPOSED AMENDMENT

The proposed change is to delete TS 3.5(5) and its associated Bases in their entirety and relocate this information to the USAR. This proposed amendment also adds a TS requirement for a Containment Tendon Testing Program as TS 5.21 consistent with that presented in Section 5.5 of Reference 10.1. This program maintains the reporting requirements of TS 3.5(5).

3.0 BACKGROUND

The proposed amendment relocates the requirements of TS 3.5(5) for testing Prestressed Concrete Containment Tendons to the USAR. This proposed amendment also adds a TS requirement for a Containment Tendon Testing Program as TS 5.21 consistent with that presented in Section 5.5 of Reference 10.1. This is acceptable since testing of containment tendons in accordance with ASME Boiler and Pressure Vessel Code, Section IX, Subsections IWE and IWL is specified in 10 CFR 50.55a. This change eliminates duplication of federal regulations and can be made without an impact on public health and safety. Therefore, this system does not meet the criteria set forth in 10 CFR 50.36(c)(2)(ii) for inclusion in the TS, and the requirements will be relocated to the USAR.

4.0 REGULATORY REQUIREMENTS AND GUIDANCE

FCS was licensed for construction prior to May 21, 1971, and at that time committed to the preliminary General Design Criteria (GDC). These preliminary design criteria are contained in the FCS USAR Appendix G.

This activity complies with FCS Design Criterion 10, "Containment," which is similar to 10 CFR 50 Appendix A GDC 16, "Containment design." FCS Design Criterion 10 states that containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity and, together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability to protect the public.

This activity also complies with FCS Design Criterion 40, "Missile Protection," which is similar to 10 CFR 50 Appendix A GDC 4, "Environmental and dynamic effects design bases." FCS Design Criterion 40 states that protection for engineered safety features shall be provided against dynamic effects and missiles that might result from plant equipment failures.

This activity also complies with FCS Design Criterion 49, "Containment Design Basis," which is similar to 10 CFR 50 Appendix A GDC 50, "Containment design basis." FCS Design Criterion 49 states that the containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the containment structure can accommodate without exceeding the design leakage rate the pressures and temperatures resulting from the largest credible energy release following a loss-of-coolant accident, including a considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of emergency core cooling systems.

This activity also complies with FCS Design Criterion 50, "NDT Requirement for Containment Material," which is similar to 10 CFR 50 Appendix A GDC 51, "Fracture prevention of containment pressure boundary." FCS Design Criterion 50 states that principal load carrying components of ferritic materials exposed to the external environment shall be selected so that their temperature under normal operating and testing conditions are not less than 30°F above nil ductility transition (NDT) temperature.

All of these FCS Design Criteria will continue to be satisfied after the change to relocate the requirements of TS 3.5(5) for testing Pre-Stressed Concrete Containment Tendons to the USAR.

5.0 TECHNICAL ANALYSIS

<u>Evaluation</u>

The proposed amendment relocates the requirements of TS 3.5(5) for testing Prestressed Concrete Containment Tendons to the USAR. This proposed amendment also adds a TS requirement for a Containment Tendon Testing Program as TS 5.21 consistent with that presented in Section 5.5 of Reference 10.1. This is acceptable since testing of containment tendons in accordance with ASME Boiler and Pressure Vessel Code, Section IX, Subsections IWE and IWL is specified in 10 CFR 50.55a. This change eliminates duplication of federal regulations and can be made without an impact on public health and safety. Therefore, this system does not meet the criteria set forth in 10 CFR

50.36(c)(2)(ii) for inclusion in the TS, and the requirements will be relocated to the USAR.

The addition of a Containment Tendon Testing Program as TS 5.21 is considered administrative since it addresses those requirements in TS 3.5(5), which is consistent with Reference 10.1. The existing reporting requirements of TS 3.5(5) are maintained in the proposed TS 5.21.

Risk Evaluation

The proposed amendment does not involve application or use of risk-informed decisions. The risk to the health and safety of the public as a result of relocating requirements of TS 3.5(5) for testing Prestressed Concrete Containment Tendons to the USAR is minimal.

6.0 REGULATORY ANALYSIS

The proposed amendment relocates the requirements of TS 3.5(5) for testing Prestressed Concrete Containment Tendons to the USAR. This proposed amendment also adds a TS requirement for a Containment Tendon Testing Program as TS 5.21 consistent with that presented in Section 5.5 of Reference 10.1. This is acceptable since testing of containment tendons in accordance with ASME Boiler and Pressure Vessel Code, Section IX, Subsections IWE and IWL is specified in 10 CFR 50.55a. This change eliminates duplication of federal regulations and can be made without an impact on public health and safety. Therefore, this system does not meet the criteria set forth in 10 CFR 50.36(c)(2)(ii) for inclusion in the TS, and the requirements will be relocated to the USAR. This complies with the regulatory requirements in FCS Design Criteria 10, 40, 49, and 50 by continuing to prevent damage to the containment structure.

In conclusion, 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) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

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

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

Response: No

The proposed change relocates requirements for testing Prestressed Concrete Containment Tendons that do not meet the criteria for inclusion in the TS set forth in 10 CFR 50.36(c)(2)(ii). The requirements for testing Prestressed Concrete Containment Tendons are being relocated from TS to the USAR, which will be maintained pursuant to 10 CFR 50.59, thereby reducing the level of regulatory control. The level of regulatory control has no impact on the probability or consequences of an accident previously evaluated. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No

The proposed change relocates requirements for testing Prestressed Concrete Containment Tendons that do not meet the criteria for inclusion in TS set forth in 10 CFR 50.36(c)(2)(ii). The change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. The change will not impose different requirements, and adequate control of information will be maintained. This change will not alter assumptions made in the safety analysis and licensing basis. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

Response: No

The proposed change relocates requirements for testing Prestressed Concrete Containment Tendons that do not meet the criteria for inclusion in TS set forth in 10 CFR 50.36(c)(2)(ii). The change will not reduce a margin of safety since the location of a requirement has no impact on any safety analysis assumptions. In addition, the relocated requirements for testing Prestressed Concrete Containment Tendons remain the same as the existing TS. Since any future changes to these requirements or the surveillance procedures will be evaluated per the requirements of 10 CFR 50.59, there will be no reduction in a margin of safety.

Based on the above, OPPD 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.

8.0 ENVIRONMENTAL CONSIDERATION

The proposed amendment relocates the requirements of TS 3.5(5) for testing Prestressed Concrete Containment Tendons to the USAR. This proposed amendment also adds a TS requirement for a Containment Tendon Testing Program as TS 5.21 consistent with that presented in Section 5.5 of Reference 10.1. This is acceptable since testing of containment tendons in accordance with ASME Boiler and Pressure Vessel Code, Section IX, Subsections IWE and IWL is specified in 10 CFR 50.55a. This change eliminates duplication of federal regulations. Therefore, this system does not meet the criteria set forth in 10 CFR 50.36(c)(2)(ii) for inclusion in the TS, and the requirements will be relocated to the USAR. The changes meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) for the following reasons:

- As demonstrated in Section 7.0, the proposed amendment does not involve a significant hazards consideration.
- The proposed amendment does not result in a significant change in the types or increase in the amounts of any effluents that may be released off-site. Also, the TS change does not introduce any new effluents or significantly increase the quantities of existing effluents. As such, the change cannot significantly affect the types or amounts of any effluents that may be released off-site.
- The proposed amendment does not result in a significant increase in individual or cumulative occupational radiation exposure. The proposed change does not result in any physical plant changes. No new surveillance requirements are anticipated as a result of these changes that would require additional personnel entry into radiation controlled areas. Therefore, the amendment has no significant affect on either 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), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

9.0 PRECEDENCE

The proposed Technical Specifications are patterned after the Improved Standard Technical Specifications as described in NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants" Reference 10.1. The NRC has approved specifications very similar to these proposed changes for the Palisades Nuclear Power Plant.

10.0 REFERENCES

10.1 NUREG-1432, "Standard Technical Specifications, Combustion Engineering Plants"

ATTACHMENT 2

Markup of
Technical Specification Pages
&
Bases Pages

~~3.0 SURVEILLANCE REQUIREMENTS~~
~~3.5 Containment Tests (Continued)~~

~~(5) Surveillance for Prestressing System~~

~~a. Sample Selection~~

~~The 210 dome tendons and 616 helical wall tendons shall be periodically inspected for symptoms of material deterioration or prestressing force reduction. Inspections shall be performed on four dome tendons, one from each layer and the control dome tendon, and ten helical wall tendons, five of each orientation including one control tendon in each orientation.~~

~~The tendons to be inspected shall be randomly selected from the tendons which have not been tested in previous surveillances, except for the control tendons which shall be included in each surveillance sample selection to develop a historical trend in order to correlate the observed data.~~

~~b. Visual Inspection~~

~~The following visual inspections shall be performed:~~

~~(i) The exterior surface of the containment shall be visually examined to detect areas of large spall, severe scaling, D-cracking in areas of 25 square feet or more, grease leakage, and other significant structural deterioration or disintegration.~~

~~(ii) For each surveillance tendon, selected in accordance with 3.5(5)a., the tendon anchorage assembly hardware shall be visually inspected for signs of abnormal material behavior or wear.~~

~~(iii) The concrete surrounding the visually inspected tendon anchorages shall be visually inspected for signs of significant structural deterioration.~~

~~(iv) The bottom grease caps of all helical wall tendons shall be visually inspected to detect grease leakage or grease cap deformations. Removal of the grease caps is not necessary for this inspection.~~

~~c. Prestress Monitoring Tests~~

~~Liftoff tests shall be performed on each tendon selected in accordance with 3.5(5)a. to monitor prestress. Additionally, the tests shall include the following:~~

~~3.0 SURVEILLANCE REQUIREMENTS~~
~~3.5 Containment Tests (Continued)~~

~~(i) Two helical wall tendons, one of each orientation, and one dome tendon, each randomly selected from their respective groups of surveillance tendons, shall be detensioned and inspected for broken or damaged wires. The control tendons shall NOT be included as tendons to be detensioned.~~

~~(ii) During retensioning, simultaneous elongation and jacking force measurements shall be made at a minimum of three approximately equally spaced levels of force between zero and the lock-off force. The two intermediate stress levels shall be as near as practical to the values shown on the initial stressing records for the respective tendon.~~

~~d. Tendon Material Tests and Inspections~~

~~One wire from each of two helical wall tendons, one of each orientation, and one dome tendon, shall be removed for the following tests and examinations:~~

~~(i) Each removed wire shall be examined over its entire length for any evidence of corrosion or other deterioration.~~

~~(ii) Tensile tests shall be made on at least three samples of each wire, one cut from each end and one cut from midlength. The samples shall be the maximum length practical for testing and the guage length for elongation shall be in accordance with ASTM E8 "Standard Test Methods for Tension Testing of Metallic Materials." The following information shall be obtained from each test:~~

~~(a) Yield Strength,~~

~~(b) Ultimate tensile strength, and~~

~~(c) Elongation at ultimate tensile strength.~~

~~**3.0 SURVEILLANCE REQUIREMENTS**~~
~~3.5 Containment Tests (Continued)~~

~~The tendons detensioned in accordance with 3.5(5)c.(i) may be the tendons from which the sample wires are removed. The control tendons shall NOT be included as tendons to be detensioned or have wires removed. In addition, all wires found to be broken shall be removed for tensile testing and visual examination.~~

~~e. Inspection of Filler Grease~~

~~A sample of sheathing filler grease from each of the sample tendons shall be taken and analyzed according to the following national standards:~~

~~(i) To determine water content, ASTM D95, "Standard Test Methods for Water in Petroleum Products and Bituminous Materials by Distillation."~~

~~(ii) To determine reserve alkalinity, ASTM D974, "Standard Test Method for Acid and Base Number by Color-Indicator Titration."~~

~~(iii) To determine the concentration of water soluble chlorides, ASTM D512, "Standard Test Methods for Chloride Ion in Water."~~

~~(iv) To determine the concentration of water soluble nitrates, ASTM D3867, "Standard Test Methods for Nitrite-Nitrate in Water."~~

~~(v) To determine the concentration of water soluble sulfides, APHA 4500-S^{2}-D. "Methylene Blue Method," Standard Methods for Examination of Water and Waste Water, Seventeenth Edition.~~

~~In addition to these tests, the amount of filler grease removed from and replaced into each surveillance tendon shall be recorded and compared to assess grease leakage within the containment structure.~~

~~f. Acceptance Criteria~~

~~(i) No evidence of significant structural deterioration of the concrete inspected in accordance with 3.5(5)b.(i) and 3.5(5)b.(iii) which may affect the structural integrity of the containment structure can be detected.~~

~~Significant structural deterioration is defined as measurable structural deterioration which, when compared with past inspections, shows strong evidence of an increase of structural deterioration which could affect the Containment's structural integrity. Evidence of cosmetic or superficial deterioration, unless determined by sound engineering judgement to be significant, is not considered to be significant structural deterioration.~~

~~No evidence of significant material degradation or corrosion of tendon anchorage hardware can be detected.~~

~~If any grease leakage is detected during visual examination of the containment exterior surface, an investigation shall be made to determine the extent of potential reduction of Containment structural integrity. An investigation shall also be made to determine which tendons could have lost the grease and whether the grease loss has adversely affected their corrosion protection.~~

~~(ii) The prestressing force measured for each tendon liftoff tested in accordance with 3.5(5)c. shall be compared with the limits predicted by USAR Fig 5.10-3. If the measured prestressing force of a selected tendon is greater than the prescribed lower limit, the tendon is acceptable.~~

~~If the measured prestressing force of a selected tendon is less than the prescribed lower limit but greater than or equal to 95% of the prescribed lower limit, the tendon shall be tensioned to a prestress value greater than the prescribed lower limit but less than 742 kips. After increasing the tendon's prestress the tendon will be considered acceptable.~~

~~3.0 SURVEILLANCE REQUIREMENTS~~
~~3.5 Containment Tests (Continued)~~

~~If the measured prestressing force of a selected tendon is less than 95% of the prescribed lower limit but greater than or equal to 90% of the prescribed lower limit, two additional tendons, one on each side of the first tendon, shall be liftoff tested. If the prestressing forces of each of the second and third tendons are greater than 95% of the prescribed lower limit, all three tendons shall be tensioned to greater than the prescribed lower limit, but less than 742 kips. After increasing the tendons' prestress, the tendons will be considered acceptable. If the prestressing force of either the second or third tendons is less than 95% of the prescribed lower limit, liftoff tests shall be performed on additional tendons to determine the cause and extent of such occurrence. This occurrence shall be considered reportable per 3.5(5)g. If the measured prestressing force of a selected tendon is less than 90% of the prescribed lower limit, the defective tendon shall be fully inspected to determine the cause and extent of such occurrence. This occurrence shall be considered reportable per 3.5(5)g.~~

~~If the average prestressing force of all measured tendons of a group (corrected for average condition) is found to be less than the prescribed lower limit, an investigation shall be performed to determine the cause and extent of such an occurrence. Such an occurrence shall be considered reportable per 3.5(5)g.~~

~~If from consecutive surveillances the average measured prestressing force of a tendon group trends at a rate which would indicate that the loss of prestress would make the average prestress of the group of tendons less than the prescribed lower limit before the next surveillance, additional liftoff tests shall be performed to determine the cause and extent of such occurrence. Such an occurrence shall be considered reportable per 3.5(5)g.~~

~~(iii) If during the detensioning and retensioning of tendons in accordance with 3.5(5)c., the elongation corresponding to a specific load differs by more than 10% from that recorded during installation of the tendons, an investigation shall be made to ensure that the difference is not related to wire failures or slippage of wires in anchorages. A difference of more than 10% shall be considered reportable per 3.5(5)g.~~

~~**3.0 SURVEILLANCE REQUIREMENTS**~~
~~3.5 Containment Tests (Continued)~~

~~(iv) The minimum acceptable ultimate tensile strength of the wire samples to be tensile tested shall be 240,000 psi with a minimum elongation of 4% in accordance with ASTM A421-65 for Type BA wire. Failure in the tensile test at strength or elongation values less than those specified shall be considered reportable per 3.5(5)g. Other conditions which indicate corrosion found by visual examination of the wire shall be considered reportable per 3.5(5)g.~~

~~(v) Results of the laboratory tests and examinations of the filler grease will be considered acceptable if the following conditions are met:~~

~~(a) Water content < 10% by weight~~

~~(b) Chlorides < 10 ppm~~

~~(c) Nitrates < 10 ppm~~

~~(d) Sulfides < 10 ppm~~

~~(e) Reserve alkalinity > 0~~
~~(Base numbers)~~

~~(f) The difference between the amount of grease injected into a tendon to replace the amount which was removed during inspection shall not exceed 5% of the net tendon sheath (duct) volume when injected at the original installation pressure.~~

~~(g) The lack of the presence of any free water.~~

~~The failure to meet any of the above conditions for the filler grease shall be considered reportable per 3.5(5)g.~~

~~g. Corrective Action and Reporting~~

~~If the above acceptance criteria are not met, an immediate investigation shall be made to determine the cause(s) and extent of the non-conformance to the criteria, and the results shall be reported to the Commission within 90 days via a special report in accordance with Technical Specification 5.9.3.~~

3.0 SURVEILLANCE REQUIREMENTS
~~3.5 Containment Tests (Continued)~~

~~h. Test Frequency~~

~~The tendon prestressing system surveillance shall be performed once every 5 years.~~

Basis

The containment is designed for an accident pressure of 60 psig.[2] While the reactor is operating, the internal environment of the containment will be air at approximately atmospheric pressure and a maximum temperature of about 120°F. With these initial conditions the temperature of the steam-air mixture at the peak accident pressure of 60 psig is 288°F.

Prior to initial operation, the containment was strength-tested at 69 psig and then was leak tested. The design objective of the pre-operational leakage rate test has been established as 0.1% by weight for 24 hours at 60 psig. This leakage rate is consistent with the construction of the containment, which is equipped with independent leak-testable penetrations and contains channels over all inaccessible containment liner welds, which were independently leak-tested during construction.

Safety analyses have been performed on the basis of a leakage rate of 0.1% of the free volume per day of the first 24 hours following the maximum hypothetical accident. With this leakage rate, a reactor power level of 1500 MWt, and with minimum containment engineered safety systems for iodine removal in operation (one air cooling and filtering unit), the public exposure would be well below 10 CFR Part 100 values in the event of the maximum hypothetical accident. (3) The performance of an integrated leakage rate test and performance of local leak rate testing of individual penetrations at periodic intervals during plant life provides a current assessment of potential leakage from the containment.

The reduced pressure (5 psig) test on the PAL is a conservative method of testing and provides adequate indication of any potential containment leakage path. The test is conducted by pressurizing between two resilient seals on each door. The test pressure tends to unseat the resilient seals which is opposite to the accident pressure that tends to seat the resilient seals. A periodic test ensures the overall PAL integrity at 60 psig.

The integrated leakage rate test (Type A test) can only be performed during refueling shutdowns.

3.0 **SURVEILLANCE REQUIREMENTS**
3.5 Containment Tests (continued)

The frequency of periodic integrated leakage rate tests is based on several major considerations: (1) There is a low probability of leaks in the liner because of the test of leak-tightness of the welds during erection and conformance of the complete containment to a low leak rate at 60 psig during pre-operational testing, which is consistent with 0.1% leakage at design basis accident conditions and absence of any significant stresses in the liner during reactor operation. (2) Periodic testing is conducted at full accident pressure, on those portions of the containment envelope that are most likely to develop leaks during reactor operation (penetrations and isolation valves). A low value (0.60 L_a) of total leakage is specified as acceptable from penetrations and isolation valves. (3) The tendon stress surveillance program provides assurance that an important part of the structural integrity of the containment is maintained. (4) A review of leakage rates obtained during past containment integrated leakage rate testing is conducted to set appropriate frequency of performance. (5) Visual inspection of the containment structure is conducted every other refueling and prior to each Integrated Leakage Rate Test.

As left leakage prior to the first startup after performing a required 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 $\leq$ 1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis.

Integrity tests of the purge isolation valves are established to identify excessive degradation of the resilient seats of these valves. Simultaneous testing of redundant purge valves from a leak test connection accessible from outside containment provides adequate testing. The testing method is identical to the Type C purge isolation valve test performed in accordance with 10 CFR Part 50, Appendix J. For leakages found to be greater than 18,000 SCCM, repairs shall be initiated to ensure these valves meet the acceptance criteria.

~~A reduction in prestressing force and changes in physical conditions are expected for the prestressing system. Allowances have been made in the reactor building design for the reduction and changes. Through comparisons between the documented inspection results and the initial quality control records, the reductions in prestress and the physical changes are trended to verify excessive reductions or changes do not occur or are detected in a timely manner to be corrected.~~

~~**3.0 SURVEILLANCE REQUIREMENTS**~~
~~3.5 Containment Tests (Continued)~~

~~The prestressing system is a necessary strength element of the plant safeguards and it is desirable to confirm that the allowances are not being exceeded. The technique chosen for surveillance is based on the rate of change of prestressing force and physical conditions so that the surveillance can either confirm that the allowances are sufficient or require maintenance before minimum levels of prestressing force or physical conditions are reached. The end anchorage concrete is needed to maintain the prestressing forces. The design investigations have concluded that the design is adequate and this has been confirmed by tests. The prestressing sequence has shown that the end anchorage concrete can withstand loads in excess of those which result when the tendons are anchored. Further, the containment building was pressure tested to 1.15 times the maximum design pressure.~~

References

(1) USAR, Section 5.9

(2) USAR, Section 5.1.1

(3) USAR, Section 14.15

5.0 ADMINISTRATIVE CONTROLS

5.7 Safety Limit Violation

5.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The unit shall be placed in at least HOT SHUTDOWN within 1 hour.

b. The Safety Limit Violations shall be reported to the corporate officer responsible for overall plant nuclear safety and the Chairperson of the Safety Audit and Review Committee (SARC) within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the Plant Review Committee. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Chairperson of the Safety Audit and Review Committee and the corporate officer responsible for overall plant nuclear safety within 14 days of the violation.

5.8 Procedures

5.8.1 Written procedures and administrative policies shall be established, implemented and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and to NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Fire Protection Program implementation; and

d. All programs specified in Specification 5.11 through 5.~~19~~*21*.

5.8.2 Temporary changes to procedures of 5.8.1 above may be provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant supervisory staff, at least one of whom holds a Senior Reactor Operator's License.

5.0 **ADMINISTRATIVE CONTROLS**

5.9 Reporting Requirements (Continued)

b. Annual Occupational Exposure Report. An annual occupational exposure report shall be submitted on or before April 30 of each year. The report shall consist of a tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions,[3/] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling outages. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

c. Monthly Operating Report. Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the U.S. Nuclear Regulatory Commission, Document Control Desk, with a copy to the appropriate Regional Office, no later than the fifteenth of each month following the calendar month covered by the report. This monthly report shall also include a statement regarding any challenges or failures to the pressurizer power operated relief valves or safety valves occurring during the subject month.

5.9.2 Reportable Event

A Licensee Event Report (LER) shall be submitted to the U.S. Nuclear Regulatory Commission for any event meeting the requirements of 10 CFR Part 50.73.

5.9.3 Special Reports

Special reports shall be submitted to the appropriate NRC Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification where appropriate:

a. In-service inspection report, reference 3.3.
b. Tendon surveillance, reference ~~3.5~~ *5.21*.
c. Containment structural tests, reference 3.5.
d. DELETED
e. DELETED
f. DELETED
g. Materials radiation surveillance specimens reports, reference 3.3.
h. DELETED
i. Post-accident monitoring instrumentation, reference 2.21
j. Electrical systems, reference 2.7(2).

3/ This tabulation supplements the requirements of § 20.2206 of 10 CFR Part 20.

5.0 **ADMINISTRATIVE CONTROLS**

5.19 Containment Leakage Rate Testing Program (Continued)

The maximum allowable primary containment leakage rate, L_a, at P_a, shall be 0.1% of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ Maximum Pathway Leakage Rate (MXPLR) for Type B and C tests and $\leq 0.75\ L_a$ for Type A tests.

b. Personnel Air Lock testing acceptance criteria are:

   (1) Overall Personnel Air Lock leakage is $\leq 0.1\ L_a$ when tested at $\geq P_a$.

   (2) For each PAL door, seal leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 5.0 psig.

c. Containment Purge Valve (PCV-742A/B/C/D) testing acceptance criterion is:

   For each Containment Purge Valve, leakage rate is < 18.000 SCCM when tested at $\geq P_a$.

d. If at any time when containment integrity is required and the total Type B and C measured leakage rate exceeds $0.60\ L_a$ Minimum Pathway Leakage Rate (MNLPR), repairs shall be initiated immediately. If repairs and retesting fail to demonstrate conformance to this acceptance criteria within 48 hours, then containment shall be declared inoperable.

The provisions of Specification 3.0.1 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of Specification 3.0.4 are applicable to the Containment Leakage Rate Testing Program.

[5.20 Bases Control Program (proposed to be added by submittal dated July 23, 2002)]

5.21 Containment Tendon Testing Program

This program provides controls for monitoring any tendon degradation in prestressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Containment Tendon Testing Program, inspection frequencies, and acceptance criteria shall be in accordance with Regulatory Guide 1.35, Revision 3, 1989.

*5.0 **ADMINISTRATIVE CONTROLS***

5.21 Containment Tendon Testing Program (Continued)

The provisions of TS 3.0.1 and TS 3.0.5 [(proposed to be added by submittal dated July 23, 2002)] *are applicable to the Containment Tendon Testing Program inspection frequencies.*

If the acceptance criteria are not met, an immediate investigation shall be made to determine the cause(s) and extent of the non-conformance to the criteria, and the results shall be reported to the Commission within 90 days via a special report in accordance with Technical Specification 5.9.3.

ATTACHMENT 3

Clean-Typed Technical Specification Pages
&
Bases Pages

NOT USED

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NOT USED

3.0 SURVEILLANCE REQUIREMENTS

Basis

The containment is designed for an accident pressure of 60 psig.[2] While the reactor is operating, the internal environment of the containment will be air at approximately atmospheric pressure and a maximum temperature of about 120°F. With these initial conditions the temperature of the steam-air mixture at the peak accident pressure of 60 psig is 288°F.

Prior to initial operation, the containment was strength-tested at 69 psig and then was leak tested. The design objective of the pre-operational leakage rate test has been established as 0.1% by weight for 24 hours at 60 psig. This leakage rate is consistent with the construction of the containment, which is equipped with independent leak-testable penetrations and contains channels over all inaccessible containment liner welds, which were independently leak-tested during construction.

Safety analyses have been performed on the basis of a leakage rate of 0.1% of the free volume per day of the first 24 hours following the maximum hypothetical accident. With this leakage rate, a reactor power level of 1500 MWt, and with minimum containment engineered safety systems for iodine removal in operation (one air cooling and filtering unit), the public exposure would be well below 10 CFR Part 100 values in the event of the maximum hypothetical accident. (3) The performance of an integrated leakage rate test and performance of local leak rate testing of individual penetrations at periodic intervals during plant life provides a current assessment of potential leakage from the containment.

The reduced pressure (5 psig) test on the PAL is a conservative method of testing and provides adequate indication of any potential containment leakage path. The test is conducted by pressurizing between two resilient seals on each door. The test pressure tends to unseat the resilient seals which is opposite to the accident pressure that tends to seat the resilient seals. A periodic test ensures the overall PAL integrity at 60 psig.

The integrated leakage rate test (Type A test) can only be performed during refueling shutdowns.

3.0 **SURVEILLANCE REQUIREMENTS**
3.5 Containment Tests (continued)

The frequency of periodic integrated leakage rate tests is based on several major considerations: (1) There is a low probability of leaks in the liner because of the test of leak-tightness of the welds during erection and conformance of the complete containment to a low leak rate at 60 psig during pre-operational testing, which is consistent with 0.1% leakage at design basis accident conditions and absence of any significant stresses in the liner during reactor operation. (2) Periodic testing is conducted at full accident pressure, on those portions of the containment envelope that are most likely to develop leaks during reactor operation (penetrations and isolation valves). A low value (0.60 La) of total leakage is specified as acceptable from penetrations and isolation valves. (3) The tendon stress surveillance program provides assurance that an important part of the structural integrity of the containment is maintained. (4) A review of leakage rates obtained during past containment integrated leakage rate testing is conducted to set appropriate frequency of performance. (5) Visual inspection of the containment structure is conducted every other refueling and prior to each Integrated Leakage Rate Test.

As left leakage prior to the first startup after performing a required 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 $\leq 1.0\ L_a$. At $\leq 1.0\ L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis.

Integrity tests of the purge isolation valves are established to identify excessive degradation of the resilient seats of these valves. Simultaneous testing of redundant purge valves from a leak test connection accessible from outside containment provides adequate testing. The testing method is identical to the Type C purge isolation valve test performed in accordance with 10 CFR Part 50, Appendix J. For leakages found to be greater than 18,000 SCCM, repairs shall be initiated to ensure these valves meet the acceptance criteria.

References

(1) USAR, Section 5.9

(2) USAR, Section 5.1.1

(3) USAR, Section 14.15

NOT USED

5.0 ADMINISTRATIVE CONTROLS

5.7 Safety Limit Violation

a.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The unit shall be placed in at least HOT SHUTDOWN within 1 hour.

b. The Safety Limit Violations shall be reported to the corporate officer responsible for overall plant nuclear safety and the Chairperson of the Safety Audit and Review Committee (SARC) within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the Plant Review Committee. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Chairperson of the Safety Audit and Review Committee and the corporate officer responsible for overall plant nuclear safety within 14 days of the violation.

5.8 Procedures

5.8.1 Written procedures and administrative policies shall be established, implemented and maintained covering the following activities:

a. The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and to NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;

c. Fire Protection Program implementation; and

d. All programs specified in Specification 5.11 through 5.21.

5.8.2 Temporary changes to procedures of 5.8.1 above may be provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant supervisory staff, at least one of whom holds a Senior Reactor Operator's License.

5.0 ADMINISTRATIVE CONTROLS

5.9 Reporting Requirements (Continued)

b. Annual Occupational Exposure Report. An annual occupational exposure report shall be submitted on or before April 30 of each year. The report shall consist of a tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions,[3/] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling outages. The dose assignment to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

c. Monthly Operating Report. Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis to the U.S. Nuclear Regulatory Commission, Document Control Desk, with a copy to the appropriate Regional Office, no later than the fifteenth of each month following the calendar month covered by the report. This monthly report shall also include a statement regarding any challenges or failures to the pressurizer power operated relief valves or safety valves occurring during the subject month.

5.9.2 Reportable Event

A Licensee Event Report (LER) shall be submitted to the U.S. Nuclear Regulatory Commission for any event meeting the requirements of 10 CFR Part 50.73.

5.9.3 Special Reports

Special reports shall be submitted to the appropriate NRC Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification where appropriate:

a. In-service inspection report, reference 3.3.
b. Tendon surveillance, reference 5.21.
c. Containment structural tests, reference 3.5.
d. DELETED
e. DELETED
f. DELETED
g. Materials radiation surveillance specimens reports, reference 3.3.
h. DELETED
i. Post-accident monitoring instrumentation, reference 2.21
j. Electrical systems, reference 2.7(2).

3/ This tabulation supplements the requirements of § 20.2206 of 10 CFR Part 20.

5.0 **ADMINISTRATIVE CONTROLS**

5.19 Containment Leakage Rate Testing Program (Continued)

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5.0 **ADMINISTRATIVE CONTROLS**

5.21 Containment Tendon Testing Program (Continued)

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