DUKE POWER COMPANY

OCONEE NUCLEAR STATION

ATTACHMENT 1

TECHNICAL SPECIFICATIONS

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3.6.6 The combined leakage rate for all penetrations and valves shall be determined in accordance with Specification 4.4.1.2. If, based on the most recent surveillance testing results the combined leakage rate exceeds the specified value and containment integrity is required then,

1) corrective action of Specification 3.6.3.c is met, or

2) repairs shall be initiated immediately and conformance with specified value shall be demonstrated within 48 hours or the reactor shall be in cold shutdown within an additional 36 hours.

3.6.7 Whenever containment integrity is required as specified in Specifications 3.6.1 and 3.6.2, the structural integrity of the reactor building(s) shall be maintained at a level consistent with the acceptance criteria identified in Specification 4.4.2.

1. If abnormal degradation of the reactor building structural integrity is indicated by the conditions in Specification 4.4.2.2.a.4,

THEN

a) Restore the reactor building(s) to the required level of structural integrity within 72 hours,

OR

b) Verify that reactor building(s) structural integrity is maintained, by performing an engineering evaluation of the reactor building(s) structural integrity, within 72 hours,

AND

c) Provide a Special Report to the Commission within 15 days in accordance with Specification 6.6.3.f,

OR

d) At the end of the 72 hour period, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. If the indicated abnormal degradation of the reactor building structural integrity, other than Action (1) above, is at a level below any other acceptance criteria of Specification 4.4.2,

THEN

a) Restore the reactor building(s) to the required level of structural integrity within 15 days,

OR

b) Verify that reactor building structural integrity is maintained by performing an engineering evaluation of the reactor building(s) structural integrity, within 15 days,

AND

c) Provide a Special Report to the Commission within 30 days in accordance with Specification 6.6.3.f,

OR

d) At the end of the 15 day period, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

BASES

The Reactor Coolant System conditions of cold shutdown assure that no steam will be formed and hence no pressure buildup in the containment if the Reactor Coolant System ruptures.

The selected shutdown conditions are based on the type of activities that are being carried out and will preclude criticality in any occurrence.

The reactor building is designed for an internal pressure of 59 psig and an external pressure 3.0 psi greater than the internal pressure. The design external pressure of 3.0 psi corresponds to a margin of 0.5 psi above the differential pressure that could be developed if the building is sealed with an internal temperature of 120 °F with a barometric pressure of 29.0 inches of Hg and the building is subsequently cooled to an internal temperature of 80 °F with a concurrent rise in barometric pressure to 31.0 inches of Hg. The weather conditions assumed here are conservative since an evaluation of National Weather Service records for this area indicates that from 1918 to 1970 the lowest barometric pressure recorded is 29.05 inches of Hg and the highest of 30.85 inches of Hg.

The reactor building is a free standing structure constructed of prestressed concrete and structural liner plate steel with no separation between the two components. The reactor building consists of a post-tensioned reinforced concrete cylinder and dome connected to, and supported by, a massive reinforced concrete foundation slab. The entire interior surface of the structure is lined with a 1/4 inch thick welded ASTM A36 steel plate to assure a high degree of leak tightness. In the concept of a prestressed concrete reactor building, the internal pressure load is balanced by the application of an opposing external pressure type load thereby assuring integrity of the structure. The Reactor Building Post-Tensioning System provides a sufficient level of prestress load on the cylinder and dome to more than balance the internal pressure so that a margin of external pressure exists beyond that required to resist the design accident pressure. The internal pressure loads on the foundation slab are resisted by both the external bearing pressure due to dead load and the strength of the reinforced concrete slab; thus, post-tensioning is not required to exert an external pressure for the foundation slab portion of the structure. Based on information provided in Regulatory Guide 1.35, the action times required to restore reactor building structural integrity are acceptable as specified in Technical Specifications 3.6.7.1 and 3.6.7.2.

Operation with a personnel or emergency hatch inoperable does not impair containment integrity since either door meets the design specifications for structural integrity and leak rate. Momentary passage through the outer door is necessary should the inner door gasket be inoperative to

install or remove auxiliary restraint beams on the inner door to allow testing of the hatch. The time limits imposed permit completion of maintenance action and the performance of a local leak rate test when required or the orderly shutdown and cooldown of the reactor. Timely corrective action for an inoperable containment isolation valve is also specified.

Penetration flow paths, except for the Reactor Building Purge flow path, may be opened on an intermittent basis under administrative controls. Per NRC Generic Letter 91-08, acceptable administrative control for opening a containment isolation valve includes (1) stationing an operator, who is in constant communication with the control room, at the valve controls, (2) instructing this operator to close the valve in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valve and that this action will prevent the release of radioactivity outside the containment.

When containment integrity is established, the limits of 10CFR100 will not be exceeded should the maximum hypothetical accident occur.

The Reactor Building purge system was designed to allow cleanup of the Reactor Building atmosphere. It is normally operated during a unit shutdown which will require entry into the Reactor Building. It is used to purge the Reactor Building with fresh air to reduce the contaminant levels within the Building atmosphere, thus reducing overall personnel exposure. At times, certain safety related functions necessitate entry into the Reactor Building prior to cold shutdown conditions. These include isolation of leaking primary coolant system valves and visual inspections following outages. Use of the purge system tends to minimize any personnel exposure while not significantly contributing to overall plant risk.

The Reactor Building Purge System is required to be isolated whenever the RCS temperature is above 250 °F and pressure is above 350 psig. The maximum pressure limit of 350 psig is based on the Oconee Unit 1 NPSH curve for RC pump operation. This will give a reasonable operating margin for the pumps while operating the purge. The LCO allows one isolation valve to be open on each penetration at or below hot shutdown for testing or maintenance.

REFERENCES

FSAR, Section 3.8

4.4.2 Reactor Building Structural Integrity

Applicability

Applies to structural integrity of the Reactor Building, specifically, the prestressed concrete cylinder and dome portions of the reactor building structure.

Objective

To define the inservice surveillance program for the Reactor Building post-tensioning system and concrete cylinder and dome.

Specification

4.4.2.1 Surveillance Intervals

Structural integrity of the prestressing tendons of the containment shall be demonstrated at the following intervals:

a. For Unit 1, the initial inspection shall be within 18 months of the Initial Reactor Building Structural Integrity Test. The inspection intervals, measured from the date of the initial inspection, shall be two years, four years and every five years thereafter.

b. For Units 2 and 3, the inspection intervals measured from the date of the Initial Structural Integrity Test shall be one year, three years and every five years thereafter.

c. Tendon surveillance may be conducted during reactor operation provided design conditions regarding loss of adjacent tendons are satisfied at all times.

4.4.2.2 Tendons

Adequacy of prestressing forces in tendons shall be demonstrated by performing the following activities:

a. Determine that a random, but representative, sample of at least eleven tendons (five hoop, three vertical, three dome) each have an observed lift-off force within the predicted limits established for each tendon. For each subsequent inspection, one tendon from each group shall be kept unchanged to develop a history and to correlate the observed data. The procedure of inspection and the tendon acceptance criteria shall be as follows:

1. If the measured prestressing force of the selected tendon in a group lies above the prescribed lower limit, the lift-off test is considered to be a positive indication of the sample tendon's acceptability.

2. If the measured prestressing force of the selected tendon in a group lies between the prescribed lower limit and 90% of the prescribed lower limit, two tendons, one on each side of this tendon, shall be checked for their prestressing forces. If the prestressing forces of these two tendons are above 95% of the prescribed lower limits for the tendons, all three tendons shall be restored to the required level of integrity, and the tendon group shall be considered acceptable. If the measured prestressing forces of any two tendons fall below 95% of the prescribed lower limits of the tendons, additional lift-off testing shall be done to detect the cause and extent of such occurrence. The conditions shall be considered as an indication of abnormal degradation of the containment structure(s). In the event of an indication of abnormal degradation, refer to Technical Specification 3.6.7.2.

3. If the measured prestressing force of any tendon lies below 90% of the prescribed lower limit, the defective tendon shall be fully investigated and additional lift-off testing shall be done so as to determine the cause and extent of such occurrence. The condition shall be considered as an indication of abnormal degradation of the containment structure. In the event of an indication of

abnormal degradation, refer to Technical Specification 3.6.7.2.

4. If the average of all measured prestressing forces for any group (corrected for average condition) is found to be less than the minimum required prestress level at anchorage location for that group, the condition shall be considered as abnormal degradation of the containment structure. In the event of an indication of abnormal degradation, refer to Technical Specification 3.6.7.1.

5. If the measured prestressing forces from consecutive surveillances for the same tendon, or tendons in a group, indicate a trend of prestress loss larger than expected and the resulting prestressing forces are likely to be less than the minimum required for the group before the next scheduled surveillance, additional lift-off testing shall be done so as to determine the cause and extent of such occurrence. The condition shall be considered as an indication of abnormal degradation of the reactor building structure. In the event of an indication of abnormal degradation, refer to Technical Specification 3.6.7.2.

b. Perform tendon detensioning, inspections, and material tests on a previously stressed tendon from each group. A randomly selected tendon from each group shall be completely detensioned in order to identify any broken or damaged wires and to determine the following conditions over the entire length of a removed tendon wire sample (this wire sample should be the broken wire if so identified):

1. Tendon wires are free of corrosion, cracks, and damage, and

2. Minimum tensile strength of 240,000 psi (guaranteed ultimate tensile strength of the wire material) exists for at least three wire samples (one from each end and one at mid-length) cut from the removed wire.

Failure to meet requirements of 4.4.2.2.b shall be considered as an indication of abnormal degradation of the reactor building structure. In the event of an indication of abnormal degradation, refer to Technical Specification 3.6.7.2.

c. Retension tendons detensioned for inspection to a force at least equal the force recorded prior to detensioning or the predicted value at the time of inspection, whichever is greater, but do not exceed 70% of the guaranteed ultimate tensile strength of the tendon wire material. Tendon seating force tolerance shall be -0 / +6%. During retensioning of these tendons, change in load versus elongation should be measured at a minimum of three approximately equally spaced levels of force between zero and the tendon seating force. If the elongation corresponding to a specific load differs by more than 10% from that recorded during the original installation, an investigation should be made to ensure that the difference is not related to wire failures or slip of wires at anchorages. This condition shall be considered as an indication of abnormal degradation of the reactor building structure. In the event of an indication of abnormal degradation, refer to Technical Specification 3.6.7.2.

d. Verify acceptability of the sheathing filler grease by assuring that:

1. No free water is present and no changes in the presence or physical appearance of the sheathing filler grease occur.
2. Amount of grease replaced does not exceed 5% of the net duct volume when injected at +/- 10% of the specified installation pressure.
3. Minimum grease coverage exists for the different parts of the anchorage system.
4. Reactor building exterior surface does not exhibit grease leakage that could affect reactor building structural integrity.
5. Chemical properties of the sheathing filler grease are within the following tolerance limits:

Water Content	0 - 10% (by dry wt.)
Chlorides	0 - 10 ppm
Nitrates	0 - 10 ppm

Sulfides	0 - 10 ppm
Reserve Alkalinity (Base Numbers)	> 50% of installed value; > 0 (for older grease)

Failure to meet requirements of 4.4.2.2.d shall be considered as an indication of potential abnormal degradation of the reactor building structure. In the event of an indication of abnormal degradation, refer to Technical Specification 3.6.7.2.

4.4.2.3 End Anchorages and Adjacent Concrete Surfaces

As an assurance of the structural integrity of the reactor building(s), tendon anchorage assembly hardware (such as bearing plates, stressing washers, wedges, and buttonheads) of all tendons selected for inspection shall be visually examined. Tendon anchorages selected for inspection shall be visually examined to the extent practical without dismantling the load bearing components of the anchorages. Bottom grease caps of all vertical tendons shall be visually inspected to detect grease leakage or grease cap deformations. The surrounding concrete should also be checked visually for indication of any abnormal condition.

Significant grease leakage, grease cap deformation or abnormal concrete condition shall be considered as an indication of abnormal degradation of reactor building structure. In the event of an indication of abnormal degradation, refer to Technical Specification 3.6.7.2.

4.4.2.4 Reactor Building Surfaces

The exterior surface of the reactor building(s) should be visually examined to detect areas of large spall, severe scaling, D-cracking in an area of 25 sq. ft. or more, other surface deterioration or disintegration, or grease leakage. Each of these conditions can be considered as evidence of abnormal degradation of structural integrity of the reactor building(s). This inspection may be performed prior to the Type A containment leakage rate test (Refer to Technical Specification 4.4.1). . In the event of an indication of abnormal degradation, refer to Technical Specification 3.6.7.2.

Bases

Provisions have been made for an inservice surveillance program intended to provide sufficient evidence that the integrity of the Reactor Building prestressed concrete containment is being preserved. This program will be conducted in accordance with the guidance of Regulatory Position C of Regulatory Guide 1.35, *Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments*, Revision 3 dated July 1990. Regulatory Guide 1.35 describes a basis acceptable to the NRC staff for developing an appropriate inservice inspection and surveillance program for ungrouted tendons in prestressed concrete containment structures of light-water-cooled reactors. The inservice surveillance program will be subject to review and revision as warranted based on studies and on results obtained for this and other prestressed concrete containments throughout the life of the plant.

6.6.3 Special Reports

Special reports shall be submitted to the Regional Administrator, Region II, 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:

a. Auxiliary Electrical Systems, Specification 3.7

b. (Not Used)

c. (Not Used)

d. Reactor Coolant System Surveillance,
   Inservice Inspection, Specification 4.2.1
   Reactor Vessel Speciment, Specification 4.2.4

e. Reactor Building Surveillance,
   Containment Leakage Tests, Specification 4.4.1

f. Structural Integrity Surveillance,
   Tendon Surveillance, Specification 3.6.7

g. (Not Used)

h. (Not Used)

DUKE POWER COMPANY

OCONEE NUCLEAR STATION

ATTACHMENT 2

CURRENT TECHNICAL SPECIFICATIONS
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Reactor Building

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3.6 REACTOR BUILDING

Applicability

Applies to the containment when the reactor is in conditions other than refueling shutdown.

Objective

To assure containment integrity during shutdown (other than refueling shutdown), startup and operation.

Specification

3.6.1 Containment integrity shall be maintained whenever all three (3) of the following conditions exist:

a. Reactor coolant pressure is 300 psig or greater
b. Reactor coolant temperature is 200°F or greater
c. Nuclear fuel is in the core

3.6.2 Containment integrity shall be maintained whenever the reactor is subcritical by less than 1% $\Delta k/k$ or whenever positive reactivity insertions are being made which would result in the reactor being subcritical by less than 1% $\Delta k/k$.

3.6.3 Exceptions to 3.6.1 and 3.6.2 shall be as follows:

a. If either the personnel or emergency hatches become inoperable, except as a result of an inoperable door gasket, the hatch shall be restored to an operable status within 24 hours, or the reactor shall be in cold shutdown within the next 36 hours.

If a hatch is inoperable due to an inoperable door gasket:

1. The remaining door of the affected hatch shall be closed and sealed. If the inner door gasket is inoperable, momentary passage (not to exceed 10 minutes for each opening) is permitted through the outer door for repair or test of the inner door, provided that the outer door gasket is leak tested within 24 hours after opening of the outer door.

2. The hatch shall be restored to operable status within seven days or the reactor shall be in cold shutdown within the next 36 hours.

b. The Reactor Building purge supply and exhaust isolation valves shall be closed except as allowed by Specification 3.6.3.b.1 and 3.6.3.b.2.

1. The Reactor Building purge system may be operated, with the supply and exhaust isolation valves open, when the Reactor Coolant System temperature is below 250°F and pressure is below 350 psig.

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2. For plant conditions when the Reactor Coolant System temperature is above 250°F and pressure is above 350 psig but the reactor is at or below hot shutdown, one Reactor Building Purge isolation valve on each penetration may be open for testing and/or maintenance per Specification 4.4.4.1 and 3.6.6.

3. For plant conditions other than contained in Specification 3.6.3.b.1, .2 above, with one or more Reactor Building purge valves open, the open valves shall be closed within one hour, or the plant shall be in hot shutdown within 12 hours and within an additional 24 hours, Reactor Coolant System temperature below 250°F and pressure below 350 psig.

c. A containment isolation valve, other than a Reactor Building Purge isolation valve, may be inoperable provided either:

1. The inoperable valve is restored to operable status within four hours.

2. The affected penetration is isolated within four hours by the use of a deactivated automatic valve secured and locked in the isolated position.[1]

3. The affected penetration is isolated within four hours by the use of a closed manual valve or blind flange.[1]

4. The reactor is in the hot shutdown condition within 12 hours and cold shutdown within 24 hours.

3.6.4 The reactor building internal pressure shall not exceed 1.5 psig or a vacuum of five inches of Hg if the reactor is critical.

3.6.5 Prior to criticality following refueling shutdown, a check shall be made to confirm that all manual containment isolation valves which should be closed are closed and tagged.

[1] Penetration flow paths (except for the Reactor Building Purge flow path) may be unisolated intermittently under administrative controls.

3.6.6 The combined leakage rate for all penetrations and valves shall be determined in accordance with Specification 4.4.1.2. If, based on the most recent surveillance testing results the combined leakage rate exceeds the specified value and containment integrity is required then,

1) corrective action of Specification 3.6.3.c is met, or

2) repairs shall be initiated immediately and conformance with specified value shall be demonstrated within 48 hours or the reactor shall be in cold shutdown within an additional 36 hours.

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BASES

The Reactor Coolant System conditions of cold shutdown assure that no steam will be formed and hence no pressure buildup in the containment if the Reactor Coolant System ruptures.

The selected shutdown conditions are based on the type of activities that are being carried out and will preclude criticality in any occurrence.

The reactor building is designed for an internal pressure of 59 psig and an external pressure 3.0 psi greater than the internal pressure. The design external pressure of 3.0 psi corresponds to a margin of 0.5 psi above the differential pressure that could be developed if the building is sealed with an internal temperature of 120°F with a barometric pressure of 29.0 inches of Hg and the building is subsequently cooled to an internal temperature of 80°F with a concurrent rise in barometric pressure to 31.0 inches of Hg. The weather conditions assumed here are conservative since an evaluation of National Weather Service records for this area indicates that from 1918 to 1970 the lowest barometric pressure recorded is 29.05 inches of Hg and the highest of 30.85 inches of Hg. ← PLACE INSERT "B" HERE

Operation with a personnel or emergency hatch inoperable does not impair containment integrity since either door meets the design specifications for structural integrity and leak rate. Momentary passage through the outer door is necessary should the inner door gasket be inoperative to install or remove auxiliary restraint beams on the inner door to allow testing of the hatch. The time limits imposed permit completion of maintenance action and the performance of a local leak rate test when required or the orderly shutdown and cooldown of the reactor. Timely corrective action for an inoperable containment isolation valve is also specified.

Penetration flow paths, except for the Reactor Building Purge flow path, may be opened on an intermittent basis under administrative controls. Per NRC Generic Letter 91-08, acceptable administrative control for opening a containment isolation valve includes (1) stationing an operator, who is in constant communication with the control room, at the valve controls, (2) instructing this operator to close the valve in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valve and that this action will prevent the release of radioactivity outside the containment.

When containment integrity is established, the limits of 10CFR100 will not be exceeded should the maximum hypothetical accident occur.

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The Reactor Building purge system was designed to allow cleanup of the Reactor Building atmosphere. It is normally operated during a unit shutdown which will require entry into the Reactor Building. It is used to purge the Reactor Building with fresh air to reduce the contaminant levels within the Reactor Building atmosphere, thus reducing overall personnel exposure. At times, certain safety related functions necessitate entry into the Reactor Building prior to cold shutdown conditions. These include isolation of leaking primary coolant system valves and visual inspections following outages. Use of the purge system tends to minimize any personnel exposure while not significantly contributing to overall plant risk.

The Reactor Building Purge System is required to be isolated whenever the RCS temperature is above 250°F and pressure is above 350 psig. The maximum pressure limit of 350 psig is based on the Oconee Unit 1 NPSH curve for RC pump operation. This will give a reasonable operating margin for the pumps while operating the purge. The LCO allows one isolation valve to be open on each penetration at or below hot shutdown for testing or maintenance.

REFERENCES

FSAR, Section 3.8

INSERT A:

3.6.7 Whenever containment integrity is required as specified in Specifications 3.6.1 and 3.6.2, the structural integrity of the reactor building(s) shall be maintained at a level consistent with the acceptance criteria identified in Specification 4.4.2.

1. If abnormal degradation of the reactor building structural integrity is indicated by the conditions in Specification 4.4.2.2.a.4,

THEN

a) Restore the reactor building(s) to the required level of structural integrity within 72 hours,

OR

b) Verify that reactor building(s) structural integrity is maintained, by performing an engineering evaluation of the reactor building(s) structural integrity, within 72 hours,

AND

c) Provide a Special Report to the Commission within 15 days in accordance with Specification 6.6.3.f,

OR

d) At the end of the 72 hour period, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. If the indicated abnormal degradation of the reactor building structural integrity, other than Action (1) above, is at a level below any other acceptance criteria of Specification 4.4.2,

THEN

a) Restore the reactor building(s) to the required level of structural integrity within 15 days,

OR

b) Verify that reactor building structural integrity is maintained by performing an engineering evaluation of the reactor

building(s) structural integrity, within 15 days,

AND

c) Provide a Special Report to the Commission within 30 days in accordance with Specification 6.6.3.f,

OR

d) At the end of the 15 day period, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

INSERT B:

The reactor building is a free standing structure constructed of prestressed concrete and structural liner plate steel with no separation between the two components. The reactor building consists of a post-tensioned reinforced concrete cylinder and dome connected to, and supported by, a massive reinforced concrete foundation slab. The entire interior surface of the structure is lined with a 1/4 inch thick welded ASTM A36 steel plate to assure a high degree of leak tightness. In the concept of a prestressed concrete reactor building, the internal pressure load is balanced by the application of an opposing external pressure type load thereby assuring integrity of the structure. The Reactor Building Post-Tensioning System provides a sufficient level of prestress load on the cylinder and dome to more than balance the internal pressure so that a margin of external pressure exists beyond that required to resist the design accident pressure. The internal pressure loads on the foundation slab are resisted by both the external bearing pressure due to dead load and the strength of the reinforced concrete slab; thus, post-tensioning is not required to exert an external pressure for the foundation slab portion of the structure. Based on information provided in Regulatory Guide 1.35, the action times required to restore reactor building structural integrity are acceptable as specified in Technical Specifications 3.6.7.1 and 3.6.7.2.

Delete this entire section - replace with new Section 4.4.2

4.4.2 Structural Integrity

Applicability

Applies to the structural integrity of the Reactor Building.

Objective

To define the inservice surveillance program for the Reactor Building.

Specification

4.4.2.1 Tendon Surveillance

For the initial surveillance program, covering the first five years of operation, nine tendons shall be selected for periodic inspection for symptoms of material deterioration or force reduction. The surveillance tendons shall consist of three horizontal tendons, one in each of three 120° sectors of the containment; three vertical tendons located at approximately 120° apart; and three dome tendons located approximately 120° apart. The following nine tendons have been selected as the surveillance tendons:

Dome	1D28
	2D28 (Units 1 & 3)
	2D29 (Unit 2)
	3D28
Horizontal	13H9
	51H9
	53H10
Vertical	23V14
	45V16
	61V16

4.4.2.1.1 Lift-Off

Lift-off readings shall be taken for all nine surveillance tendons.

4.4.2.1.2 Wire Inspection and Testing

One surveillance tendon of each directional group shall be relaxed and one wire from each relaxed tendon shall be removed as a sample and visually inspected for corrosion or pitting. Tensile tests shall also be performed on a minimum of three specimens taken from the ends and middle of each of the three wires. The specimens shall be the maximum length acceptable for the test apparatus to be used and shall include areas representative of significant corrosion or pitting.

After the wire removal, the tendons shall be retensioned to the stress level measured at the lift-off reading and then checked by a final lift-off reading.

Should the inspection of one of the wires reveal any significant corrosion (pitting or loss of area), further inspection of the other two sets in that directional group will be made to determine the extent of the corrosion and its significance to the load-carrying capability of the structure. The sheathing filler will be sampled and inspected for changes in physical appearance.

Wire samples shall be selected in such a manner that with the third inspection, wires from all nine surveillance tendons shall have been inspected and tested.

4.4.2.2 Inspection Intervals and Reports

For Unit 1, the initial inspection shall be within 18 months of the initial Reactor Building Structural Integrity Test. The inspection intervals, measured from the date of the initial inspection, shall be two years, four years and every five years thereafter or as modified based on experience. For Units 2 and 3 the inspection intervals measured from the date of the initial structural test shall be one year, three years and every five years thereafter or as modified based on experience. Tendon surveillance may be conducted during reactor operation provided design conditions regarding loss of adjacent tendons are satisfied at all times.

A quantitative analytical report covering results of each inspection shall be submitted to the Commission within 90 days of completion, and shall especially address the following conditions, should they develop.

a. Broken wires.

b. The force-time trend line for any tendon, when extrapolated, that extends beyond either the upper or lower bounds of the predicted design band.

c. Unexpected changes in corrosion conditions or sheathing filler properties.

Bases

Provisions have been made for an in-service surveillance program, covering the first several years of the life of the unit, intended to provide sufficient evidence to maintain confidence that the integrity of the Reactor Building is being preserved. This program consists of tendon, tendon anchorage and liner plate surveillance. The first year tendon anchorage and liner plate surveillance programs have been successfully completed.

To accomplish these programs, the following representative tendon groups have been selected for surveillance:

Horizontal - Three 120° tendons comprising one complete hoop system below grade

Vertical - Three tendons spaced approximately 120° apart.

Dome - Three tendons spaced approximately 120° apart.

~~The inspection during this initial period of at least one wire from each of the nine surveillance tendons (one wire per group per inspection) is considered sufficient representation to detect the presence of any wide spread tendon corrosion or pitting conditions in the structure. This program will be subject to review and revision as warranted based on studies and on results obtained for this and other prestressed concrete reactor buildings during this period of time.~~

4.4.2 Reactor Building Structural Integrity

Applicability

Applies to structural integrity of the Reactor Building, specifically, the prestressed concrete cylinder and dome portions of the reactor building structure.

Objective

To define the inservice surveillance program for the Reactor Building post-tensioning system and concrete cylinder and dome.

Specification

4.4.2.1 Surveillance Intervals

Structural integrity of the prestressing tendons of the containment shall be demonstrated at the following intervals:

a. For Unit 1, the initial inspection shall be within 18 months of the Initial Reactor Building Structural Integrity Test. The inspection intervals, measured from the date of the initial inspection, shall be two years, four years and every five years thereafter.

b. For Units 2 and 3, the inspection intervals measured from the date of the Initial Structural Integrity Test shall be one year, three years and every five years thereafter.

c. Tendon surveillance may be conducted during reactor operation provided design conditions regarding loss of adjacent tendons are satisfied at all times.

4.4.2.2 Tendons

Adequacy of prestressing forces in tendons shall be demonstrated by performing the following activities:

a. Determine that a random, but representative, sample of at least eleven tendons (five hoop, three vertical, three dome) each have an observed lift-off force within the predicted limits established for each tendon. For each subsequent inspection, one tendon from each group shall be kept unchanged to develop a history and to correlate the observed data. The procedure of inspection and the tendon acceptance criteria shall be as follows:

1. If the measured prestressing force of the selected tendon in a group lies above the prescribed lower limit, the lift-off test is considered to be a positive indication of the sample tendon's acceptability.

2. If the measured prestressing force of the selected tendon in a group lies between the prescribed lower limit and 90% of the prescribed lower limit, two tendons, one on each side of this tendon, shall be checked for their prestressing forces. If the prestressing forces of these two tendons are above 95% of the prescribed lower limits for the tendons, all three tendons shall be restored to the required level of integrity, and the tendon group shall be considered acceptable. If the measured prestressing forces of any two tendons fall below 95% of the prescribed lower limits of the tendons, additional lift-off testing shall be done to detect the cause and extent of such occurrence. The conditions shall be considered as an indication of abnormal degradation of the containment structure(s). In the event of an indication of abnormal degradation, refer to Technical Specification 3.6.7.2.

3. If the measured prestressing force of any tendon lies below 90% of the prescribed lower limit, the defective tendon shall be fully investigated and additional lift-off testing shall be done so as to determine the cause and extent of such occurrence. The condition shall be considered as an indication of abnormal degradation of the containment structure. In the event of an indication of

abnormal degradation, refer to Technical Specification 3.6.7.2.

4. If the average of all measured prestressing forces for any group (corrected for average condition) is found to be less than the minimum required prestress level at anchorage location for that group, the condition shall be considered as abnormal degradation of the containment structure. In the event of an indication of abnormal degradation, refer to Technical Specification 3.6.7.1.

5. If the measured prestressing forces from consecutive surveillances for the same tendon, or tendons in a group, indicate a trend of prestress loss larger than expected and the resulting prestressing forces are likely to be less than the minimum required for the group before the next scheduled surveillance, additional lift-off testing shall be done so as to determine the cause and extent of such occurrence. The condition shall be considered as an indication of abnormal degradation of the reactor building structure. In the event of an indication of abnormal degradation, refer to Technical Specification 3.6.7.2.

b. Perform tendon detensioning, inspections, and material tests on a previously stressed tendon from each group. A randomly selected tendon from each group shall be completely detensioned in order to identify any broken or damaged wires and to determine the following conditions over the entire length of a removed tendon wire sample (this wire sample should be the broken wire if so identified):

1. Tendon wires are free of corrosion, cracks, and damage, and

2. Minimum tensile strength of 240,000 psi (guaranteed ultimate tensile strength of the wire material) exists for at least three wire samples (one from each end and one at mid-length) cut from the removed wire.

Failure to meet requirements of 4.4.2.2.b shall be considered as an indication of abnormal degradation of the reactor building structure. In the event of an indication of abnormal degradation, refer to Technical Specification 3.6.7.2.

c. Retension tendons detensioned for inspection to a force at least equal the force recorded prior to detensioning or the predicted value at the time of inspection, whichever is greater, but do not exceed 70% of the guaranteed ultimate tensile strength of the tendon wire material. Tendon seating force tolerance shall be -0 / +6%. During retensioning of these tendons, change in load versus elongation should be measured at a minimum of three approximately equally spaced levels of force between zero and the tendon seating force. If the elongation corresponding to a specific load differs by more than 10% from that recorded during the original installation, an investigation should be made to ensure that the difference is not related to wire failures or slip of wires at anchorages. This condition shall be considered as an indication of abnormal degradation of the reactor building structure. In the event of an indication of abnormal degradation, refer to Technical Specification 3.6.7.2.

d. Verify acceptability of the sheathing filler grease by assuring that:

1. No free water is present and no changes in the presence or physical appearance of the sheathing filler grease occur.
2. Amount of grease replaced does not exceed 5% of the net duct volume when injected at +/- 10% of the specified installation pressure.
3. Minimum grease coverage exists for the different parts of the anchorage system.
4. Reactor building exterior surface does not exhibit grease leakage that could affect reactor building structural integrity.
5. Chemical properties of the sheathing filler grease are within the following tolerance limits:

Water Content	0 - 10% (by dry wt.)
Chlorides	0 - 10 ppm
Nitrates	0 - 10 ppm

Sulfides	0 - 10 ppm
Reserve Alkalinity (Base Numbers)	> 50% of installed value; > 0 (for older grease)

Failure to meet requirements of 4.4.2.2.d shall be considered as an indication of potential abnormal degradation of the reactor building structure. In the event of an indication of abnormal degradation, refer to Technical Specification 3.6.7.2.

4.4.2.3 End Anchorages and Adjacent Concrete Surfaces

As an assurance of the structural integrity of the reactor building(s), tendon anchorage assembly hardware (such as bearing plates, stressing washers, wedges, and buttonheads) of all tendons selected for inspection shall be visually examined. Tendon anchorages selected for inspection shall be visually examined to the extent practical without dismantling the load bearing components of the anchorages. Bottom grease caps of all vertical tendons shall be visually inspected to detect grease leakage or grease cap deformations. The surrounding concrete should also be checked visually for indication of any abnormal condition.

Significant grease leakage, grease cap deformation or abnormal concrete condition shall be considered as an indication of abnormal degradation of reactor building structure. In the event of an indication of abnormal degradation, refer to Technical Specification 3.6.7.2.

4.4.2.4 Reactor Building Surfaces

The exterior surface of the reactor building(s) should be visually examined to detect areas of large spall, severe scaling, D-cracking in an area of 25 sq. ft. or more, other surface deterioration or disintegration, or grease leakage. Each of these conditions can be considered as evidence of abnormal degradation of structural integrity of the reactor building(s). This inspection may be performed prior to the Type A containment leakage rate test (Refer to Technical Specification 4.4.1). . In the event of an indication of abnormal degradation, refer to Technical Specification 3.6.7.2.

Bases

Provisions have been made for an inservice surveillance program intended to provide sufficient evidence that the integrity of the Reactor Building prestressed concrete containment is being preserved. This program will be conducted in accordance with the guidance of Regulatory Position C of Regulatory Guide 1.35, *Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments*, Revision 3 dated July 1990. Regulatory Guide 1.35 describes a basis acceptable to the NRC staff for developing an appropriate inservice inspection and surveillance program for ungrouted tendons in prestressed concrete containment structures of light-water-cooled reactors. The inservice surveillance program will be subject to review and revision as warranted based on studies and on results obtained for this and other prestressed concrete containments throughout the life of the plant.

6.6.3 Special Reports

Special reports shall be submitted to the Regional Administrator, ~~Region II,~~ 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:

a. Auxiliary Electrical Systems, Specification 3.7

b. (Not Used)

c. (Not Used)

d. Reactor Coolant System Surveillance,
   Inservice Inspection, Specification 4.2.1
   Reactor Vessel Specimen, Specification 4.2.4

e. Reactor Building Surveillance,
   Containment Leakage Tests, Specification 4.4.1

f. Structural Integrity Surveillance,
   Tendon Surveillance, Specification ~~4.4.2.2~~ 3.6.7

g. (Not Used)

h. (Not Used)

Amendment No. 194 (Unit 1)
Amendment No. 194 (Unit 2)
Amendment No. 191 (Unit 3)

ATTACHMENT 3

TECHNICAL JUSTIFICATION

Technical Justification

The proposed amendment to Oconee Technical Specifications completely rewrites Specification 4.4.2 in order for Oconee to implement a prestressed concrete containment surveillance program which is consistent with Regulatory Guide 1.35. This change to the current Reactor Building Post-Tensioning System surveillance program will allow Oconee to obtain new inspection data. This new data will provide improved knowledge of the current state of the Reactor Building Post-Tensioning System and also enhance Duke's ability to predict the future state of the system. The proposed changes to Specification 3.6 will assure appropriate station response to abnormal degradation of the containment structure as indicated by conditions specified in proposed Specification 4.4.2. The new Specification 3.6.7, as proposed, will establish a Limiting Condition for Operation and required actions for the structural integrity of the Oconee reactor buildings. An editorial change to Specification 6.6.3 is being performed to reference the relocated tendon surveillance reporting requirements.

Oconee UFSAR Section 3.8.1.7.6 describes the current surveillance program for the Reactor Building Post-Tensioning System. The program consists of periodic inspections of nine pre-designated surveillance tendons - three horizontal, three vertical, and three dome - on each unit. The specific tendons are currently tabulated in the Specification 4.4.2.1. These tendons are inspected in accordance with the requirements of Specification 4.4.2 for symptoms of material deterioration and excessive loss of prestress force. The program assesses the condition and functional capability of the Reactor Building Post-Tensioning System. Evaluation of the results of the inspection program provides an opportunity to take timely corrective action for any indication of structural deterioration of the reactor building(s); however, a surveillance program based upon pre-designated samples is inherently limited in scope. Additionally, experience has shown that long term wear on the pre-designated sample tendons, caused by repetitive surveillance activities, can adversely affect the quality of observed surveillance data.

Since initial plant licensing, Oconee has based its Reactor Building Post-Tensioning System inspection program on pre-

designated surveillance tendons. Use of pre-designated tendons as the basis for a Reactor Building Post-Tensioning System surveillance program is custom to Oconee. General industry practice is to employ a program which is consistent with Regulatory Guide 1.35 and which requires random sampling of the tendon population for inspection. Data obtained from inspection of the pre-designated surveillance tendons has been tainted by tendon force fluctuation (either increase or decrease), wire breakage, and stressing washer thread damage caused by repetitive tendon detensioning / retensioning activities. To date, surveillance tendons 2D28 of Unit 2 and 51H9 of Unit 1 have incurred sufficient surveillance-induced damage so as to preclude their use in future inspections. Damage to tendon 2D28 occurred during the second Unit 2 inspection (November 1978). Damage to tendon 51H9 occurred during the sixth Unit 1 inspection (February 1993).

Current Specification 4.4.2 requires submittal of a quantitative analytical report which describes the results of each Reactor Building Post-Tensioning System inspection. In a letter dated October 11, 1995, Duke submitted this report for the Oconee Unit 3 Reactor Building Post-Tensioning System sixth surveillance. By letter dated January 19, 1996, the NRC requested additional information regarding this report. Duke's response to the NRC request for additional information was dated March 14, 1996, and included a commitment to revise Specification 4.4.2 to allow the implementation of a prestressed concrete containment surveillance program consistent with the guidance provided in Regulatory Guide 1.35. The time frame for submittal of this proposed amendment to the Oconee Technical Specifications is intended to support its implementation during the next scheduled Reactor Building Post-Tensioning System inspection in April 1997 (Oconee Unit 1 Outage EOC 17).

Regulatory Guide 1.35, Revision 3, dated July 1990, provides detailed NRC guidance on the inservice inspection of ungrouted tendons in prestressed concrete containments. With respect to identification of surveillance tendons, the Regulatory Guide details an acceptable technique for random sample selection that is based upon a percentage (4%) of the Reactor Building tendon population in each group (vertical, hoop, and dome) for inspections conducted at 1, 3, and 5 years. Provided that these inspections yield acceptable results, the sample size may be reduced to 2% of each tendon group for subsequent inspections. In both the 4% and 2% sampling sizes, upper and lower limits on the sample size from any tendon group are specified. Regulatory Position

1.5 of Regulatory Guide 1.35 provides relief, with respect to tendon liftoff force comparison, for sites with more than one plant. However, the relaxed surveillance intervals for tendon liftoff force comparison for sites with two plants, as depicted in Figure 1 of Regulatory Guide 1.35, are not readily applicable to a three-plant site such as Oconee.

Containment post-tensioning system surveillance conducted in accordance with the proposed amendment to TS 4.4.2 will utilize the reduced tendon population sample size (2%) but will not take advantage of the relaxed surveillance intervals, with respect to tendon liftoff force comparison, specified for sites with more than one plant. Based on the following reasons, Duke believes that inspection, including lift-off force testing, of a tendon population sample size of 2% with a surveillance frequency of once every five years for each of the three Oconee plants will assure an effective inservice inspection and surveillance program.

1) Oconee will perform its seventh post-tensioning system surveillance (twenty-fifth year) of Oconee Unit 1 (Outage EOC 17). Seventh surveillances of Oconee Units 2 and 3 (twenty-sixth year and twenty-fifth year, respectively) will be performed during refueling outages scheduled to occur in 1999. Given the maturity of the Oconee plants, significant changes in the level of prestress, as determined by comparison of tendon liftoff forces between surveillance intervals, will no longer occur.

2) In the event that evaluation of inspection data indicates adverse degradation of the post-tensioning system, Regulatory Positions 7.1.2, 7.1.3, 7.1.4 and 7.1.6 of Regulatory Guide 1.35 mandate that additional liftoff testing be performed. This additional liftoff testing will determine the cause and extent of any adverse degradation, thereby assuring that a sufficient amount of surveillance data is obtained to verify the structural integrity of the reactor building(s).

3) The containments are identical in all aspects such as size, tendon system, design, materials of construction, and method of construction. In addition, the Unit 1 Initial Structural Integrity Test was performed within two years of the Unit 2 test, and the Unit 2 test was performed within two years of the Unit 3 test. There is no unique situation that may subject any of these three containments to a different potential for structural or tendon deterioration.

In summary, three factors form the bases for the proposed amendment to Oconee Technical Specifications:

1) Improvement in the confidence in Reactor Building Post-Tensioning System inspection data through random sampling of the tendon population,

2) Damage to pre-designated surveillance tendon 51H9 of Unit 1 incurred during the sixth surveillance of the Unit 1 Reactor Building Post-Tensioning system, and

3) Duke's commitment to implement a prestressed concrete containment surveillance program consistent with the guidance of Regulatory Guide 1.35.

The proposed technical specification amendment will move Oconee into a surveillance program which is consistent with both an accepted industry practice and a published regulatory position. The adoption of Regulatory Guide 1.35 as a basis for the periodic inspection of the reactor building(s) will assure that sufficient data is obtained to demonstrate that the structural integrity of the prestressed concrete containment is maintained, and that any adverse trends in the behavior of the prestressed concrete containment are identified and acted upon in a timely manner. Regulatory Position 2 of Regulatory Guide 1.35 delineates the method of determining sample size and emphasizes random sampling of the tendon population as opposed to using the same pre-designated surveillance tendons for each surveillance interval. Data obtained in this manner will be more representative of the tendon population in general and will therefore afford a higher degree of confidence (i.e., increase in safety) in the continued structural integrity of the prestressed concrete containment.

Concurrent with the distribution of Revision 3 of Regulatory Guide 1.35, the NRC also provided a sample Specification for utilities to use as guidance for pursuing related Technical Specification amendments. This sample Technical Specification illustrates an acceptable method of incorporating the provisions of Regulatory Guide 1.35 into an individual plant's Technical Specifications. Duke used this sample Technical Specification to prepare the proposed changes to Oconee Technical Specifications 3.6 and 4.4.2.

For Specification 3.6, the content is unchanged from the sample Specification with the exception of the requirement to shut down to hot shutdown within 6 hours and to proceed to cold shutdown in additional 30 hours. To reflect the

current licensing basis, the current Oconee Technical Specification 3.0 guidance for this action is appropriate. Current Specification 3.0 requirements are to shut down to hot shutdown in 12 hours and to proceed to cold shutdown in an additional 24 hours. Other changes were made to ensure reference to correct Specification references, and to enhance the human performance aspects of the sample Specification.

The UFSAR has been reviewed for any impacts resulting from this proposed Technical Specification. UFSAR Section 3.8.1.7.6 will be changed to reflect the new tendon surveillance activities as described in this amendment.

Rule Change to 10 CFR 50.55a

The NRC has finalized changes to 10 CFR 50.55a which invoke the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, Subsections IWE and IWL. This rule will require that all nuclear utilities perform inservice inspections of containments in accordance with the 1992 Edition of the ASME Code, through 1992 Addenda (with specified exceptions and limitations). For Oconee, this new rule will require that surveillance inspections be performed in a manner similar to that specified in Regulatory Guide 1.35 and as proposed in this requested Technical Specifications amendment. Additionally, the final rule change to 10 CFR 50.55a, as approved by the NRC, only allows for utilities to credit Reactor Building Post-Tensioning System inspections performed in accordance with Regulatory Guide 1.35 requirements to satisfy the IWL provisions for the expedited examination period (5 years from the effective date of the final rule). Implementation of the IWL requirements will require significant logistical preparations by Duke. As a result, these activities can not be completed by the next scheduled surveillance interval in April 1997 (Oconee Unit 1 Outage EOC 17). Therefore, Oconee will implement Regulatory Guide 1.35 requirements under this proposed Technical Specifications amendment as an interim measure until final implementation of the new 10 CFR 50.55a rule.

ATTACHMENT 4

NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

Pursuant to 10 CFR 50.92, Duke Power Company has determined that the proposed amendment involves No Significant Hazards Considerations. This determination was made by applying the NRC established standards contained in regulation 10 CFR 50.92. These standards assure that changes to the operation of Oconee Nuclear Station in accordance with the proposed amendment consider the following:

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

No. The proposed amendment to Oconee Technical Specifications involves the implementation of an enhanced surveillance program for the reactor building prestressed concrete containment and the assurance of appropriate station response to abnormal degradation of the containment structure. The proposed change will move Oconee into a surveillance program which is consistent with accepted industry practice and a published NRC regulatory position. The adoption of Regulatory Guide 1.35 as a basis for the periodic inspection of the reactor building prestressed concrete containment and clearly defined station response to any indication of structural deterioration will assure acquisition of sufficient data to demonstrate that structural integrity is maintained and, if necessary, appropriate compensatory action(s) are taken. By assuring that any adverse trends in the behavior of the prestressed concrete containment are identified and acted upon in a timely manner, this change does not increase the probability or consequences of an accident previously evaluated.

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

No. The proposed amendment to Oconee Technical Specifications involves the implementation of an enhanced surveillance program for the reactor building prestressed concrete containment and the assurance of appropriate station response to abnormal degradation of the containment structure. By adopting Regulatory Guide 1.35 as a basis for the surveillance inspection program for the reactor building prestressed concrete containment and clearly defining

required station response to any indication of structural deterioration, sufficient data will be obtained to demonstrate that structural integrity is being maintained and that any adverse behavioral trends are identified and acted upon in a timely manner. Therefore, the proposed amendment does not create the possibility of any type of accident: new, different, or previously evaluated.

3) Will the change involve a significant reduction in a margin of safety?

No. Margin of safety is associated with confidence in the ability of the fission product barriers (i.e., fuel and fuel cladding, Reactor Coolant System pressure boundary, and containment structure) to limit the level of radiation dose to the public. The proposed Technical Specifications amendment will move Oconee into a surveillance program which is consistent with accepted industry practice and a published regulatory position. By ensuring more timely identification of, and response to, any adverse trend in the behavior of the reactor building prestressed concrete containment, continued maintenance of the structural integrity is enhanced. Therefore, the ability of the containment structure to perform the intended function of protecting the public from radiation dose is further assured, and no reduction in any existing margin of safety will occur.

ATTACHMENT 5

ENVIRONMENTAL ASSESSMENT

The proposed amendment to Oconee Technical Specifications will allow the implementation of an enhanced surveillance program for the reactor building prestressed concrete containment and assure appropriate station response to abnormal degradation of the containment structure. The enhanced surveillance program and clearly defined station response to any indication of structural deterioration will help assure the ability of the containment structure(s) to prevent any adverse radiation impact upon the environment. Additionally, it has been determined that the requested change to the existing surveillance program involves:

1) No significant hazards consideration,

2) No significant change in the types, or significant increase in the amounts, of any effluents that may be released offsite, and

3) No significant increase in individual or cumulative occupational radiation exposures.

Therefore, the proposed amendment to Oconee Technical Specifications meets the criteria of 10 CFR 51.22(c)(9) for categorical exclusion from an environmental assessment.