DUKE POWER COMPANY

OCONEE NUCLEAR STATION

ATTACHMENT 2

TECHNICAL SPECIFICATIONS

Remove Page	Insert Page
iv	iv
3.6-2a	3.6-3
3.6-3	3.6-4
3.6-3a	3.6-5
------	3.6-6
4.4-14	4.4-14
4.4-15	4.4-15
4.4-16	4.4-16
------	4.4-16a
------	4.4-16b
------	4.4-16c
6.6-5	6.6-5

Section		Page
3.10	GAS STORAGE TANK AND EXPLOSIVE GAS MIXTURE	3.10-1
3.11	(Not Used)	3.11-1
3.12	REACTOR BUILDING POLAR CRANE AND AUXILIARY HOIST	3.12-1
3.13	SECONDARY SYSTEM ACTIVITY	3.13-1
3.14	SNUBBERS	3.14-1
3.15	CONTROL ROOM PRESSURIZATION AND FILTERING SYSTEM AND PENETRATION ROOM VENTILATION SYSTEMS	3.15-1
3.16	HYDROGEN PURGE SYSTEM	3.16-1
3.17	(NOT USED)	
3.18	STANDBY SHUTDOWN FACILITY	3.18-1
4	SURVEILLANCE REQUIREMENTS	4.0-1
4.0	SURVEILLANCE STANDARDS	4.0-1
4.1	OPERATIONAL SAFETY REVIEW	4.1-1
4.2	STRUCTURAL INTEGRITY OF ASME CODE CLASS 1, 2 AND 3 COMPONENTS	4.2-1
4.3	TESTING FOLLOWING OPENING OF SYSTEM	4.3-1
4.4	REACTOR BUILDING	4.4-1
4.4.1	Containment Leakage Tests	4.4-1
4.4.2	Reactor Building Structural Integrity	4.4-14
4.4.3	Hydrogen Purge System	4.4-17
4.4.4	Reactor Building Purge System	4.4-20
4.5	EMERGENCY CORE COOLING SYSTEMS AND REACTOR BUILDING COOLING SYSTEMS PERIODIC TESTING	4.5-1
4.5.1	Emergency Core Cooling Systems	4.5-1
4.5.2	Reactor Building Cooling Systems	4.5-4
4.5.3	Containment Heat Removal Capability	4.5-6
4.5.4	Penetration Room Ventilation System	4.5-7
4.5.5	Low Pressure Injection System Leakage	4.5-9
4.6	EMERGENCY POWER PERIODIC TESTING	4.6-1
4.7	REACTOR CONTROL ROD SYSTEM TESTS	4.7-1
4.7.1	Control Rod Trip Insertion Time	4.7-1
4.7.2	Control Rod Program Verification	4.7-2
4.8	MAIN STEAM STOP VALVES	4.8-1

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 post-tensioned reinforced concrete structure. The Reactor Building consists of a vertical cylinder supported by a reinforced concrete foundation slab and supporting a shallow domed roof. The entire interior surface of the structure is covered with a 0.25 inch thick welded steel liner plate. The Reactor Building Post-Tensioning system serves to provide a counter-acting force to the internal pressure. The internal pressure load on the foundation slab is resisted by the foundation reaction due to dead load and by the strength of the reinforcing. Based on information provided in Regulatory Guide 1.35, the action times required to restore the 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 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 $^{\circ}$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.

<u>REFERENCES</u>

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.

Objective

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

Specification

4.4.2.1 Inspection Intervals

The inspection intervals to demonstrate the structural integrity of the reactor building shall be as follows:

a. For Unit 1, the inspection interval, as measured from 7/1/91, shall be every five years thereafter.

b. For Unit 2, the inspection interval, as measured from 11/1/94, shall be every five years thereafter.

c. For Unit 3, the inspection interval, as measured from 6/1/95, shall be every five years thereafter.

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

e. Inspection intervals in Specification 4.4.2.1 (a), (b), and (c) may be modified in accordance with the requirements of ASME Section XI, Subsection IWL.

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 group. 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 reactor building(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 reactor building. 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 reactor building. 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. 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 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. 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 varying levels of force. The following table provides levels of force, pressure, and elongation at which measurements should be taken:

	Force (Kips)	Pressure (psi)	Elongation (In)
PTF			
Step 1			
Step 2			
LOF			
OSF			

Where:
Total Elongation (actual) = (LOF-PTF) Elongation
PTF - Pretensioning Force necessary to bring the tendon into a slightly stressed condition to remove slack and seat the buttonheads.
Step 1-2 - An intermediate force approximately equally spaced between PTF and LOF.
LOF - Lock Off Force at which the tendon is seated on the shims.
OSF - Overstress Force at which the maximum elongation is measured.

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. 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 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)	$\geq$ 50% of installed value; $\geq$ 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. 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. Top and 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 the reactor building. 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 inspection program intended to provide sufficient evidence that the integrity of the Reactor Building 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 reactor buildings of light-water-cooled reactors. The inservice inspection 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 throughout the life of the plant.

Prior to implementation of Regulatory Guide 1.35 methodology in accordance with this specification, Reactor Building Post Tensioning System surveillances were performed by examining specific, pre-designated test tendons. Therefore, this specification conservatively identifies the date of the last surveillance performed for each unit under the superseded Technical Specification 4.4.2, and measures the periodicity of future inspections from these dates.

Seating forces for all tendons were documented at the time of installation, thus providing one data point. A second point will be obtained from data obtained during the initial tendon surveillance for each unit. The data from the initial surveillance is considered reliable since any error due to tensioning and retensioning had not been introduced. This data will be averaged on a per unit basis and used in the trend analysis along with new data obtained from the new proposed surveillance program in accordance with Regulatory Guide 1.35.

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 3

CURRENT TECHNICAL SPECIFICATIONS

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Section | Page

Section		Page
3.10	GAS STORAGE TANK AND EXPLOSIVE GAS MIXTURE	3.10-1
3.11	(Not Used)	3.11-1
3.12	REACTOR BUILDING POLAR CRANE AND AUXILIARY HOIST	3.12-1
3.13	SECONDARY SYSTEM ACTIVITY	3.13-1
3.14	SNUBBERS	3.14-1
3.15	CONTROL ROOM PRESSURIZATION AND FILTERING SYSTEM AND PENETRATION ROOM VENTILATION SYSTEMS	3.15-1
3.16	HYDROGEN PURGE SYSTEM	3.16-1
3.17	(NOT USED)	
3.18	STANDBY SHUTDOWN FACILITY	3.18-1
4	SURVEILLANCE REQUIREMENTS	4.0-1
4.0	SURVEILLANCE STANDARDS	4.0-1
4.1	OPERATIONAL SAFETY REVIEW	4.1-1
4.2	STRUCTURAL INTEGRITY OF ASME CODE CLASS 1, 2 AND 3 COMPONENTS	4.2-1
4.3	TESTING FOLLOWING OPENING OF SYSTEM	4.3-1
4.4	REACTOR BUILDING	4.4-1
4.4.1	Containment Leakage Tests	4.4-1
4.4.2	Structural Integrity	4.4-14
4.4.3	Hydrogen Purge System	4.4-17
4.4.4	Reactor Building Purge System	4.4-20
4.5	EMERGENCY CORE COOLING SYSTEMS AND REACTOR BUILDING COOLING SYSTEMS PERIODIC TESTING	4.5-1
4.5.1	Emergency Core Cooling Systems	4.5-1
4.5.2	Reactor Building Cooling Systems	4.5-4
4.5.3	Containment Heat Removal Capability	4.5-6
4.5.4	Penetration Room Ventilation System	4.5-7
4.5.5	Low Pressure Injection System Leakage	4.5-9
4.6	EMERGENCY POWER PERIODIC TESTING	4.6-1
4.7	REACTOR CONTROL ROD SYSTEM TESTS	4.7-1
4.7.1	Control Rod Trip Insertion Time	4.7-1
4.7.2	Control Rod Program Verification	4.7-2
4.8	MAIN STEAM STOP VALVES	4.8-1

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 250oF 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 post-tensioned reinforced concrete structure. The Reactor Building consists of a vertical cylinder supported by a reinforced concrete foundation slab and supporting a shallow domed roof. The entire interior surface of the structure is covered with a 0.25 inch thick welded steel liner plate. The Reactor Building Post-Tensioning system serves to provide a counter-acting force to the internal pressure. The internal pressure load on the foundation slab is resisted by the foundation reaction due to dead load and by the strength of the reinforcing. Based on information provided in Regulatory Guide 1.35, the action times required to restore the 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.~~

A 104/104/101
11/6/81

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 Inspection Intervals

The inspection intervals to demonstrate the structural integrity of the reactor building shall be as follows:

a. For Unit 1, the inspection interval, as measured from 7/1/91, shall be every five years thereafter.

b. For Unit 2, the inspection interval, as measured from 11/1/94, shall be every five years thereafter.

c. For Unit 3, the inspection interval, as measured from 6/1/95, shall be every five years thereafter.

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

e. Inspection intervals in Specification 4.4.2.1 (a), (b), and (c) may be modified in accordance with the requirements of ASME Section XI, Subsection IWL.

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 reactor building(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 reactor building. 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 reactor building. 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. 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 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. 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 varying levels of force. The following table provides levels of force, pressure, and elongation at which measurements should be taken:

	Force (Kips)	Pressure (psi)	Elongation (In)
PTF			
Step 1			
Step 2			
LOF			
OSF			

Where:
Total Elongation (actual) = (LOF-PTF) Elongation
PTF - Pretensioning Force necessary to bring the tendon into a slightly stressed condition to remove slack and seat the buttonheads.
Step 1-2 - An intermediate force approximately equally spaced between PTF and LOF.
LOF - Lock Off Force at which the tendon is seated on the shims.
OSF - Overstress Force at which the maximum elongation is measured.

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. 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 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. In the event of an indication of abnormal degradation, refer to Technical Specification 3.6.7.2.

4.4.2.3 <u>End Anchorages and Adjacent Concrete Surfaces</u>

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. Top and 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. In the event of an indication of abnormal degradation, refer to Technical Specification 3.6.7.2.

4.4.2.4 <u>Reactor Building Surfaces</u>

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 inspection program intended to provide sufficient evidence that the integrity of the Reactor Building 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 reactor buildings of light-water-cooled reactors. The inservice inspection 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 throughout the life of the plant.

Prior to implementation of Regulatory Guide 1.35 methodology in accordance with this specification, Reactor Building Post Tensioning System surveillances were performed by examining specific, pre-designated test tendons. Therefore, this specification conservatively identifies the date of the last surveillance performed for each unit under the superseded Technical Specification 4.4.2, and measures the periodicity of future inspections from these dates.

Seating forces for all tendons were documented at the time of installation, thus providing one data point. A second point will be obtained from data obtained during the initial tendon surveillance for each unit. The data from the initial surveillance is considered reliable since any error due to tensioning and retensioning had not been introduced. This data will be averaged on a per unit basis and used in the trend analysis along with new data obtained from the new proposed surveillance program in accordance with Regulatory Guide 1.35.

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)