Attachment 4

OCONEE NUCLEAR STATION
UNIT 1

PROPOSED TECHNICAL SPECIFICATION REVISION

INSERVICE INSPECTION

7911250 089

4 SURVEILLANCE REQUIREMENTS

4.0 SURVEILLANCE STANDARDS

Applicability

Applies to surveillance requirements which relate to tests, calibrations and inspections necessary to assure that the quality of structures, systems and components is maintained and that operation is within the safety limits and limiting conditions for operation.

Objective

To specify minimum acceptable surveillance requirements.

Specification

4.0.1 Surveillance of structures, systems, components and parameters shall be as specified in the various subsections to this Technical Specification section, Section 4.0, except as permitted by Technical Specifications 4.0.2 and 4.0.3 below.

4.0.2 Minimum surveillance frequencies, unless specified otherwise, may be adjusted as follows to facilitate test scheduling:

Specified Frequency	Maximum Allowable Interval Between Surveillances
Five times per week	2 days
Two times per week	5 days
Weekly	10 days
Bi-Weekly	20 days
Monthly	45 days
Bi-Monthly	90 days
Quarterly	135 days
Semiannually	270 days
Annually	18 months

4.0.3 If conditions exist such that surveillance of an item is not necessary to assure that operation is within the safety limits and limiting conditions for operation, surveillance need not be performed if such conditions continue for a length of time greater than the specified surveillance interval. Surveillance waived as a result of this specification shall be performed prior to returning to conditions for which the surveillance is necessary to assure that operation is within safety limits and limiting conditions for operation.

4.0.4 Inservice testing of ASME Code Class 1, 2, and 3 pumps and valves for Oconee 1 shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10CFR50 Section 50.55a(g)(4) to the extent practicable within the limitations of design, geometry and materials of construction of the components.

4.2 REACTOR COOLANT SYSTEM SURVEILLANCE

Applicability

Applies to the surveillance of the Reactor Coolant System pressure boundary.

Objective

To assure the continued integrity of the Reactor Coolant System pressure boundary.

Specification

4.2.1 For Oconee 1, inservice examination of ASME Code Class 1, 2, and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10CFR50, Section 50.55a(g)(4) to the extent practicable within the limitations of design, geometry and materials of construction of the components.

4.2.2 For Oconee 2 and 3, inspections of components shall be made in accordance with the methods and intervals indicated in IS-242 and IS-261 of Section XI of the ASME Boiler and Pressure Vessel Code, 1970, including 1970 winter addenda, except as follows:

IS-261 Item	Component	Exception
1.4	Primary Nozzle to Vessel Welds	1 RC outlet nozzle to be inspected after approximately 3 1/3 years operation. 2nd RC outlet nozzle to be inspected after approx. 6 2/3 yrs. operation. 4 RC inlet nozzles and 2 core flooding nozzles to be inspected at or near end of interval
3.3	Primary Nozzle to Safe End Welds	Not Applicable
4.3	Valve Pressure Retaining Bolting Larger than 2"	Not Applicable
6.1	Valve Body Welds	Not Applicable
6.3	Valve to Safe End Welds	Not Applicable
6.6	Integrally Welded Valve Supports	Not Applicable
6.7	Valve Supports & Hangers	Not Applicable

4.2.3 The structural integrity of the Reactor Coolant System boundary shall be maintained at the level required by the original acceptance standards throughout the life of the station. Any evidence, as a result of the tests outlined in Section XI of the code, that defects have developed or grown, shall be investigated, including evaluation of comparable areas of the Reactor Coolant System.

4.2.4 The results of the Inservice Inspections performed pursuant to Specifications 4.2.1, 4.2.2, and 4.2.3 shall be reported to the Commission within 90 days of completion.

4.2.5 To assure the structural integrity of the reactor internals throughout the life of the unit, the two sets of main internals bolts (connecting the core barrel to the core support shield and to the lower grid cylinder) shall remain in place and under tension. This will be verified by visual inspection to determine that the welded bolt locking caps remain in place. All locking caps will be inspected after hot functional testing and whenever the internals are removed from the vessel during a refueling or maintenance shutdown. The core barrel to core support shield caps will be inspected each refueling shutdown.

4.2.6 At approximately three-year intervals, the bore and keyway of each reactor coolant pump flywheel shall be subjected to an in-place, volumetric examination. Whenever maintenance or repair activities necessitate flywheel removal, a surface examination of exposed surfaces and a complete volumetric examination shall be performed, if the interval measured from the previous such inspection is greater than 6 2/3 years.

4.2.7 For Unit 1 and Unit 2, a B Type vessel specimen capsule shall be withdrawn after one year of operation and an A Type capsule shall be withdrawn after 11, 17, and 22 years of operation. The withdrawal schedules may be modified to coincide with those refueling outages or unit shutdowns most closely approaching the withdrawal schedule. Specimens thus withdrawn shall be tested in accordance with ASTM-E-185-70. For Unit 3, a B Type vessel specimen capsule shall be withdrawn after one year of operation and an A Type capsule shall be withdrawn after 7, 14, and 17 years of operation. The withdrawal schedules may be modified to coincide with those refueling outages or unit shutdowns most closely approaching the withdrawal schedule. Specimens thus withdrawn shall be tested in accordance with ASTM-E-185-72. The results of these examinations shall be reported to the Commission within 90 days of completion of testing.

4.2.8 For Unit 1, Cycle 3 operation, the surveillance capsules will be removed from the reactor vessel and the provisions of Specification 4.2.7 will be revised prior to Cycle 4 operation. For Unit 2, Cycle 2 operation, the surveillance capsules will be removed from the reactor vessel and the provisions of Specification 4.2.7 will be revised prior to Cycle 3 operation. For Unit 3, Cycle 1 operation, the surveillance capsules will be removed from the reactor vessel for a portion of the cycle and the provisions of Specification 4.2.7 will be revised prior to Cycle 2 operation.

4.2.9 During the first two refueling periods, two reactor coolant system piping elbows shall be ultrasonically inspected along their longitudinal welds (4 inches beyond each side) for clad bonding and for cracks in both the clad and base metal. The elbows to be inspected are identified in B&W Report 1364 dated December 1970.

4.2.10 To assure that reactor internals vent valves are not opening during operation, all vent valves will be inspected during each refueling outage to confirm that no vent valve is stuck open and that each valve operates freely.

Bases

The surveillance program has been developed to comply with the applicable edition of Section XI and addenda of the ASME Boiler and Pressure Vessel Code, Inservice Inspection of Nuclear Reactor Coolant Systems, as required by 10CFR 50.55 a to the extent practicable within limitations of design, geometry and materials of construction for Unit 1 and with Section XI, 1970, including 1970 winter addenda edition for Units 2 and 3. The program places major emphasis on the area of highest stress concentrations and on areas where fast neutron irradiation might be sufficient to change material properties.

The reactor vessel specimen surveillance program for Unit 1 and Unit 2 is based on equivalent exposure times of 1.8, 19.8, 30.6 and 39.6 years. The contents of the different type of capsules are defined below.

A Type	B Type
Weld Material	HAZ Material
HAZ Material	Baseline Material
Baseline Material	

For Unit 3, the Reactor Vessel Surveillance Program is based on equivalent exposure times of 1.8, 13.3, 26.7, and 30.0 years. The specimens have been selected and fabricated as specified in ASTM-E-185-72.

Early inspection of Reactor Coolant System piping elbows is considered desirable in order to reconfirm the integrity of the carbon steel base metal when explosively clad with sensitized stainless steel. If no degradation is observed during the two annual inspections, surveillance requirements will revert to Section XI of the ASME Boiler and Pressure Vessel Code.

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
	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, shoudl 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 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 time and other prestressed concrete reactor buildings during this period of time.

4.5 EMERGENCY CORE COOLING SYSTEMS AND REACTOR BUILDING COOLING SYSTEM PERIODIC TESTING

4.5.5 Emergency Core Cooling Systems

Applicability

Applies to periodic testing requirements for the Emergency Core Cooling Systems.

Objective

To verify that the Emergency Core Cooling Systems are operable.

Specification

4.5.1.1 System Tests

4.5.1.1.1 High Pressure Injection System

a. Annually, a system test shall be conducted to demonstrate that the system is operable. A test signal will be applied to demonstrate actuation of the High Pressure Injection System for emergency core cooling operation.

b. The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly; all appropriate pump breakers shall have opened or closed and all valves shall have completed their travel.

4.5.1.1.2 Low Pressure Injection System

a. Annually, a system test shall be conducted to demonstrate that the system is operable. The test shall be performed in accordance with the procedure summarized below:

   (1) A test signal will be applied to demonstrate actuation of the Low Pressure Injection System for emergency core cooling operation.

   (2) Verification of the engineered safety features function of the Low Pressure Service Water System which supplies cooling water to the low pressure coolers shall be made to demonstrate operability of the coolers.

b. The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly; all appropriate pump breakers shall have opened or closed, and all valves shall have completed their travel.

4.5.1.1.3 Core Flooding System

a. Annually, a system test shall be conducted to demonstrate proper operation of the system. During pressurization of the Reactor Coolant System, verification shall be made that the check and isolation valves in the core flooding tank discharge lines operate properly.

(b) The test will be considered satisfactory if control board indication of core flood tank level verifies that all valves have opened.

4.5.1.2 Component Tests

4.5.1.2.1 Pumps

Quarterly, for Units 2 and 3, the high pressure and low pressure injection pumps shall be started and operated to verify proper operation. Acceptable performance will be indicated if the pump starts, operates for 15 minutes, and the discharge pressure and flow are within $\pm$ 10 percent of a point on the pump head curve. (Figure 4.5.1-1 and 4.5.1-2)

4.5.1.2.2 Valves - Power Operated

a. Quarterly, for Units 2 and 3, each Engineered Safety Features valve in the Emergency Core Cooling Systems and each Engineered Safety Features valve associated with emergency core cooling in the Low Pressure Service Water System shall be tested to verify operability.

b. The acceptable performance of each power-operated valve will be that motion is indicated upon actuation by appropriate signals.

c. Annually, low pressure injection pump discharge (engineered safety features) valves, low pressure injection discharge throttling valves, and low pressure injection discharge header crossover valves shall be cycled manually to verify the manual operability of these power-operated valves.

Bases

The Emergency Core Cooling Systems are the principle reactor safety features in the event of a loss of coolant accident. The removal of heat from the core provided by these systems is designed to limit core damage.

The High Pressure Injection System under normal operating conditions has one pump operating. At least once per month, operation is rotated to another high pressure injection pump. This verifies that the high pressure injection pumps are operable.

The requirements of the Low Pressure Service Water System for cooling water are more severe during normal operation than under accident conditions. Rotation of the pump in operation on a monthly basis verifies that two pumps are operable.

The low pressure injection pumps are tested singularly for operability by opening the borated water storage tank outlet valves and the bypass valves in the borated water storage tank fill line. This allows water to be pumped from the borated water storage tank through each of the injection lines and back to the tank.

(2) Verification of the engineered safety features function of the Low Pressure Service Water System which supplies coolant to the reactor building coolers shall be made to demonstrate operability of the coolers.

(b) The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly, the appropriate pump breakers have completed their travel, fans are running at half speed, LPSW flow through each cooler exceeds 1400 GPM and air flow through each fan exceeds 40,000 CFM.

4.5.2.2 Component Tests

4.5.2.2.1 Pumps

Quarterly, for Units 2 and 3, the reactor building spray pumps shall be started and operated to verify proper operation. Acceptable performance will be indicated if the pump starts, operates for 15 minutes, and the measured discharge pressure and flow results in a point above the pump head curve. (Figure 4.5.2-1)

4.5.2.2.2 Valves

Quarterly, for Units 2 and 3, each engineered safety features valve in the Reactor Building Spray and Reactor Building Cooling System and each engineered safety features valve associated with Reactor Building cooling in the Low Pressure Service Water System shall be tested to verify that it is operable.

Bases

The Reactor Building Coolant System and Reactor Building Spray System are designed to remove heat in the containment atmosphere to control the rate of depressurization in the containment. The peak transient pressure in the containment is not affected by the two heat removal systems. Hence, the basis for the spray pump flow acceptance test is the flow rate required during recirculation (1,000 gpm).

The delivery capability of one reactor building spray pump at a time can be tested by opening the valve in the line from the borated water storage tank, opening the corresponding valve in the test line, and starting the corresponding pump. Pump discharge pressure and flow indication demonstrate performance.

With the pumps shut down and the borated water storage tank outlet closed, the reactor building spray injection valves can each be opened and closed by operator action. With the reactor building spray inlet valves closed, low pressure air or fog can be blown through the test connections of the reactor building spray nozzles to demonstrate that the flow paths are open.

The equipment, piping, valves, and instrumentation of the Reactor Building Cooling System are arranged so that they can be visually inspected. The cooling units and associated piping are located outside the secondary concrete shield. Personnel can enter the Reactor Building during power operations to inspect and maintain this equipment. The service water piping and valves outside the Reactor Building are inspectable at all times. Operational tests and inspections will be performed prior to initial startup.

4.9 EMERGENCY FEEDWATER PUMP PERIODIC TESTING

Applicability

Applies to the periodic testing of the turbine-driven emergency feedwater pump for Units 2 and 3.

Objective

To verify that the emergency feedwater pump and associated valves are operable.

Specification

4.9.1 Test

Quarterly, the turbine-driven emergency feedwater pump shall be operated on recirculation to the upper surge tank for a minimum of one hour.

4.9.2 Acceptance Criteria

These tests shall be considered satisfactory if control board indication and visual observation of the equipment demonstrates that all components have operated properly.

Bases

The quarterly testing frequency is sufficient to verify that the turbine-driven emergency feedwater pump is operable. Verification of correct operation is made both from the control room instrumentation and direct visual observation of the pump.

REFERENCES

(1) FSAR, Section 10.2.2

(2) FSAR, Section 14.1.2.8.3