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RECIP.NAME RECIPIENT AFFILIATION
DENTON,H.R. Office of Nuclear Reactor Regulation
REID,R.W. Operating Reactors Branch 4

SUBJECT: Forwards proposed Tech Specs revisions re Surveillance Stds to suppl 761001 & 770708 submittals.

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DUKE POWER COMPANY
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May 30, 1979

WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

TELEPHONE: AREA 704
373-4083

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. R. W. Reid, Chief
Operating Reactors Branch #4

Re: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287

Dear Sir:

In response to the February 16, 1976 Federal Register Notice concerning 10CFR 50.55a and NRC letter dated April 26, 1976, a review of the inservice inspection program requiremetns with regard to Oconee was conducted.

Technical Specifications for Unit 1 were submitted on October 1, 1976 and for Units 2 and 3 on July 8, 1977. Recent review of these submittals and of the inservice inspection program itself has indicated the need to resubmit Technical Specifications to reflect the current program.

My letter of May 15, 1979 provided in one document, a composite inservice inspection program for Oconee Nuclear Station. The purpose of this letter is to provide in one document proposed technical specifications required to reflect the current inservice inspection program. This submittal supplements the earlier submittals of October 1, 1976 and July 8, 1977. In this regard, no license fees are provided.

Very truly yours,

William O. Parker, Jr.

William O. Parker, Jr.

RLG:scs
Attachment

A001
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7906070238

WILLIAM O. PARKER, JR., being duly sworn, states that he is Vice President of Duke Power Company; that he is authorized on the part of said Company to sign and file with the Nuclear Regulatory Commission this request for amendment of the Oconee Nuclear Station Technical Specifications, Appendix A to Facility Operating Licenses DPR-38, DPR-47, and DPR-55; and that all statements and matters set forth therein are true and correct to the best of his knowledge.

William O. Parker, Jr., Vice President

Subscribed and sworn to before me this 30th day of May, 1979.

Notary Public

My Commission Expires:

February 15, 1982

Oconee Nuclear Station

Proposed Technical Specification Revision

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 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 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. The following exception applies:

Component	Exception
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.

4.2.2 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.

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

4.2.4 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.

4.2.5 At approximately three-year intervals, the bore and keyway of each reactor coolant pump flywheel shall be subjected to an inplace, 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.

4.2.6 The reactor vessel material irradiation surveillance specimens removed from Units 1, 2 and 3 reactor vessels in 1976 shall be installed, irradiated in and withdrawn from the Crystal River Unit 3 reactor vessel in accordance with the schedule shown in Table 4.2-1. Following withdrawal of each capsule listed in Table 4.2.-1; Duke Power Company shall be responsible for testing the specimens in those capsules and submitting a report of test results in accordance with 10CFR50, Appendix H.

4.2.7 The licensee shall submit a report or application for license amendment to the NRC within 90 days after the occurrence of the following: After March 13, 1978, any time that Crystal River Unit No. 3 fails to maintain a cumulative reactor utilization factor of greater than 45%.

The report shall provide justification for continued operation of Oconee Nuclear Station Units 1, 2 and 3 with the reactor vessel surveillance program conducted at Crystal River Unit No. 3 or the application for license amendment shall propose an alternate program for conduct of the reactor vessel surveillance program.

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 10CFR50.55(a) to the extent practicable within limitations of design, geometry and materials of construction. 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 number of reactor vessel specimens and the frequencies for removing and testing these specimens are provided to assure compliance with the requirements of Appendix H to 10CFR Part 50.

For the purpose of Technical Specification 4.2.7. Cumulative reactor utilization factor is defined as: {(Cumulative thermal megawatt hours since attainment of commercial operation at 100% power) x 100} ÷ {(licensed thermal power) x (cumulative hours since attainment of commercial operation at 100% power)}. The definition of Regulatory Guide 1.16, Revision 4 (August 1975) applies for the term "commercial operation".

Table 4.2-1

OCONEE NUCLEAR STATION CAPSULE ASSEMBLY
WITHDRAWAL SCHEDULE AT CRYSTAL RIVER UNIT NO. 3

Capsule Designation	Insertion	Withdrawal
OCI-A	End of 1st Cycle	End of 7th Cycle
OCI-B	End of 7th Cycle	End of 16th Cycle
OCI-C	End of 2nd Cycle	End of 11th Cycle
OCI-D	End of 9th Cycle	End of 18th Cycle
OCII-A	End of 1st Cycle	End of 2nd Cycle
OCII-B	End of 4th Cycle	End of 9th Cycle
OCII-D	End of 9th Cycle	End of 18th Cycle
OCII-E	End of 1st Cycle	End of 9th Cycle
OCII-F	End of 9th Cycle	End of 18th Cycle
OCIII-B	End of 1st Cycle	End of 2nd Cycle
OCIII-C	End of 2nd Cycle	End of 7th Cycle
OCIII-D	End of 1st Cycle	End of 9th Cycle
OCIII-E	End of 5th Cycle	End of 18th Cycle
OCIII-F	End of 11th Cycle	End of 20th Cycle

NOTE: OCI - Capsules are from Unit No. 1
OCII - Capsules are from Unit No. 2
OCIII - Capsules are from Unit No. 3

4.3 TESTING FOLLOWING OPENING OF SYSTEM

Applicability

Applies to test requirements for Reactor Coolant System Integrity.

Objective

To assure Reactor Coolant System integrity prior to return to criticality following normal opening, modification, or repair.

Specification

4.3.1 When Reactor Coolant System repairs or modifications have been made, these repairs or modifications shall be inspected and tested to meet all applicable code requirements prior to the reactor being made critical.

4.3.2 Following any opening of the Reactor Coolant System, it shall be leak tested at not less than 2285 psig prior to the reactor being made critical.

4.3.3 The limitations of Specification 3.1.2 shall apply.

Bases

Repairs or modifications made to the Reactor Coolant System are inspectable and testable under applicable codes, such as USAS B31.7, code for pressure piping, Nuclear Power Piping dated February, 1968, and as corrected for errata under date of June, 1968, and pursuant to 10CFR 50.55a, the ASME Boiler and Pressure Vessel Code, Section XI.

REFERENCES

FSAR, Section 4.

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

4.5.1 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.

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.

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.

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 damge.

Testing the manual operability of power-operated valves in the Low Pressure Injection System gives assurance that flow can be established in a timely manner even if the capability to operate a valve from the control room is lost.

With the reactor shut down, the valves in each core flooding line are checked for operability by reducing the Reactor Coolant System Pressure until the indicated level in the core flood tanks verify that the check and isolation valves have opened.

Reference

(1) FSAR, Section 6

4.5.2 Reactor Building Cooling Systems

Applicability

Applies to testing of the Reactor Building Cooling Systems.

Objective

To verify that the Reactor Building Cooling Systems are operable.

Specification

4.5.2.1 System Tests

4.5.2.1.1 Reactor Building Spray System

a. Annually, a system test shall be conducted to demonstrate proper operation of the system. A test signal will be applied to demonstrate actuation of the Reactor Building Spray System (except for reactor building inlet valves to prevent water entering nozzles). Water will be circulated from the borated water storage tank through the reactor building spray pumps and returned through the test line to the borated water storage tank.

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

4.5.2.1.2 Reactor Building Cooling System

a. Annually, a system test shall be conducted to demonstrate proper operation of the system. The test shall be performed in accordance with the procedure summarized below:

(1) A test signal will be applied to actuate the Reactor Building Cooling System for reactor building cooling operation.

(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.

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.

The reactor building fans are normally operated periodically, constituting the test that these fans are operable.

REFERENCE

(1) FSAR, Section 6

Specification deleted