NRC FORM 195 (2-76)

U.S. NUCLEAR REGULATORY COMMISSION

DOCKET NUMBER 50-270/287

FILE NUMBER

NRC DISTRIBUTION FOR PART 50 DOCKET MATERIAL

TO: Mr. Edson G. Case

FROM: Duke Power Company
Charlotte, North Carolina
William O. Parker, Jr.

DATE OF DOCUMENT 7/8/77

DATE RECEIVED 7/15/77

☒ LETTER ☐ NOTORIZED
☒ ORIGINAL ☒ UNCLASSIFIED
☐ COPY

PROP

INPUT FORM

NUMBER OF COPIES RECEIVED 3 SIGNED

DESCRIPTION

DO NOT REMOVE

ACKNOWLEDGED

(1-P)

ENCLOSURE

License No. DPR-39 & DPR-47..Appl for Amend: tech spec proposed change concerning the inservice inspection program...... notorized 7/8/77.....

(17-P)

PLANT NAME:
Oconee Units 2 & 3
RJL 7/15/77

FOR ACTION/INFORMATION

SAFETY			ENVIRONMENTAL
ASSIGNED AD:			ASSIGNED AD: V. MOORE (LTR)
✓ BRANCH CHIEF: (7)	SCHWENCER		BRANCH CHIEF:
PROJECT MANAGER:			PROJECT MANAGER:
LICENSING ASSISTANT:			LICENSING ASSISTANT:
			B. HARLESS

INTERNAL DISTRIBUTION

✓ REG FILES	SYSTEMS SAFETY	PLANT SYSTEMS	SITE SAFETY & ENVIRON ANALYSIS
✓ NRC PDR	HEINEMAN	TEDESCO	DENTON & MULLER
✓ I & E (2)	SCHROEDER	BENAROYA	CRUTCHFIELD
✓ OELD		LAINAS	
✓ GOSSICK & STAFF	ENGINEERING	IPPOLITO	ENVIRO TECH.
HANAUER	KNIGHT	F. ROSA	ERNST
MIPC	BOSNAK		BALLARD
CASE	SIHWELL	OPERATING REACTORS	YOUNGBLOOD
BOYD	PAWLICKI	✓ STELLO	
		✓ EISENHUT	
PROJECT MANAGEMENT	REACTOR SAFETY	✓ SHAO	~~SITE TECH.~~
SKOVHOLT	ROSS	✓ BAER	GAMMILL (2)
P. COLLINS	NOVAK	✓ BUTLER	
HOUSTON	ROSZTOCZY	✓ GRIMES	SITE ANALYSIS
MELTZ	✓ CHECK		VOLLMER
HELTEMES			BUNCH
SK	AT&I		✓ J. COLLINS
	SALTZMAN		KREGER
	RUTBERG		

EXTERNAL DISTRIBUTION

✓ LPDR: WABHALLA SC		
✓ TIC	✓ NSIC	
NAT LAB		
REG IV (J. HANCHETT)		
✓ 16 CYS ACRS SENT CATEGORY B		

CONTROL NUMBER

Appl 771960243

B

NRC FORM 195 (2-76)

DUKE POWER COMPANY
POWER BUILDING
422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28242

July 8, 1977

WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

TELEPHONE: AREA 704
373-4083

Mr. Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. A. Schwencer, Chief
Operating Reactor Branch #1

Reference: Oconee Units 2 and 3
Docket Nos. 50-270, -287

Dear Sir:

The provisions of 10CFR50.55a §(g)(5)(ii) require that the licensee apply to the Commission for amendment of the Technical Specifications should conflicts exist between the Technical Specifications and the revised inservice inspection program required by 10CFR55a. Your letter of November 30, 1976 defined a conflict to exist in those cases in which the requirements of the regulation are more restrictive than the requirements of the current Technical Specifications. Although we disagree with this definition (It is our feeling that a conflict exists when the Technical Specifications prohibit compliance with the regulations or vice versa), it is agreed that the most efficient method to eliminate existing or potential conflicts is to substitute generalized language in the Technical Specifications in place of existing inservice inspection and testing requirements.

The beginning of the next forty month intervals for Oconee Unit 2 and 3 are January 9 and April 16, 1978, respectively. In order to facilitate the implementation of the new inservice inspection program, both Oconee Unit 2 and 3 will begin the programs together on January 9, 1978. This program will utilize the ASME Section XI Code through the Summer 1975 addenda. Details of this program will be provided by October 9, 1977.

Pursuant to 10CFR50.55a and 10CFR50.90, the attached proposed Technical Specification amendments are requested. These amendments incorporate generalized language for inservice inspection; delete specific testing requirements for pumps and valves from the specifications; delete the requirement for testing of components prior to removing redundant components from service since increased testing will provide assurance of component operability for these short periods of time; and remove some requirements which have been satisfied and are no longer required.

Very truly yours,

William O. Parker, Jr.

William O. Parker, Jr.

MST:ge

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 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

William O. Parker, Jr., Vice President

Subscribed and sworn to before me this 8th day of July, 1977.

Vivian B. Rakkins

Notary Public

My Commission Expires:
Feb. 15, 1982

3.3.6 Exceptions to 3.3.5 shall be as follows:

(a) Both core flooding tanks shall be operational above 800 psig.

(b) Both motor-operated valves associated with the core flooding tanks shall be fully open above 800 psig.

(c) One pressure instrument channel and one level instrument channel per core flood tank shall be operable above 800 psig.

(d) One reactor building cooling fan and associated cooling unit shall be permitted to be out of service for seven days provided both reactor building spray pumps and associated spray nozzle headers are in service at the same time.

Bases

The requirements of Specification 3.3 assure that, before the reactor can be made critical, adequate engineered safety features are operable. Two high pressure injection pumps and two low pressure injection pumps are specified. However, only one of each is necessary to supply emergency coolant to the reactor in the event of a loss-of-coolant accident. Both core flooding tanks are required as a single core flood tank has insufficient inventory to reflood the core.(1)

The borated water storage tanks are used for two purposes:

(a) As a supply of borated water for accident conditions.

(b) As a supply of borated water for flooding the fuel transfer canal during refueling operation.(2)

Three-hundred and fifty thousand (350,000) gallons of borated water (a level of 46 feet in the BWST) are required to supply emergency core cooling and reactor building spray in the event of a loss-of-core cooling accident. This amount fulfills requirements for emergency core cooling. The borated water storage tank capacity of 388,000 gallons is based on refueling volume requirements. Heaters maintain the borated water supply at a temperature to prevent freezing. The boron concentration is set at the amount of boron required to maintain the core 1 percent subcritical at 70°F without any control rods in the core. This concentration is 1,338 ppm boron while the minimum value specified in the tanks is 1,800 ppm boron.

The spray system utilizes common suction lines with the low pressure injection system. If a single train of equipment is removed from either system, the other train must be assured to be operable in each system.

It has been shown for the worst design basis loss-of-coolant accident (a 14.1 ft^2 hot leg break) that the reactor building design pressure will not be exceeded with one spray and two coolers operable. Therefore, a maintenance period of seven days is acceptable for one reactor building cooling fan and its associated cooling unit. (3)

In the event that the need for emergency core cooling should occur, functioning of one train (one high pressure injection pump, one low pressure injection pump, and both core flooding tanks) will protect the core and in the event of a main coolant loop severence, limit the peak clad temperature to less than 2,300°F and the metal-water reaction to that representing less than 1 percent of the clad.

Three low pressure service water pumps serve Oconee Units 1 and 2 and two low pressure service water pumps serve Oconee Unit 3. There is a manual cross-connection on the supply headers for Units 1, 2, and 3. One low pressure service water pump per unit is required for normal operation. The normal operating requirements are greater than the emergency requirements following a loss-of-coolant accident.

REFERENCES

(1) FSAR, Section 14.2.2.3
(2) FSAR, Section 9.5.2
(3) FSAR, Supplement 13
(4) FSAR, Section 6.4

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.

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 Reactor coolant pump flywheels shall be examined as follows. An inplace volumetric examination of the bore and keyway shall be performed at approximately three-year intervals coinciding with ASME Section XI inservice inspection examinations. Additionally, a surface examination of all exposed surfaces and a complete ultrasonic volumetric examination of the flywheel shall be conducted at approximately ten-year intervals coinciding with ASME Section XI Inservice inspection examinations.

4.2.6 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.

4.2.7 For Unit 1, Cycle 3 operation, the surveillance capsules will be removed from the reactor vessel and the provisions of Specification 4.2.6 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 4.2.6 will be revised 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 4.2.6 will be revised prior to Cycle 2 operation.

4.2.8 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 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 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.

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^{o} tendons comprising one complete hoop system below grade

Vertical - Three tendons spaced approximately 120^{o} apart.

Dome - Three tendons spaced approximately 120^{o} 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.

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 shallbbe 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 damage.

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

DELETE

DELETE

(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

DELETE

DELETE