ATTACHMENT 1

DUKE POWER COMPANY
OCONEE NUCLEAR STATION

Proposed Technical Specification Revision
Refueling Outage Surveillance Requirements

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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
Refueling Outage	22 months, 15 days

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.

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

4.1.1 The frequency and type of surveillance required for Reactor Protective System and Engineered Safety Feature Protective System instrumentation shall be as stated in Table 4.1-1.

4.1.2 Equipment and sampling test shall be performed as detailed in Tables 4.1-2 and 4.1-3.

4.1.3 Using the Incore Instrumentation System, a power map shall be made to verify expected power distribution at periodic intervals not to exceed ten effective full power days.

Bases

Failures such as blown instrument fuses, defective indicators, and faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator action. Comparision of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear systems, when the unit is in operation, the minimum checking frequency stated in deemed adequate for reactor system instrumentation.

Calibration is performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels amplifiers are calibrated (during steady-state operating conditions) when indicated neutron power exceeds core thermal power by more than two percent. During non-steady-state operation, the nuclear flux channels amplifiers are calibrated daily to compensate for instrumentation drift and changing rod patterns and core physics parameters. Calibration checks are also performed following significant changes in core conditions (power level and control rod positions) in order to assure that the core thermal power indication during non-steady-state operations does not exceed the indicated neutron power by more than the tolerance (4% FP) assumed in the safety analysis for significant duration (e.g., 4 hours).

Channels subject only to "drift" errors induced within the instrumentation itself can tolerate longer intervals between calibrations. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed at the intervals specified.

Substantial calibration shifts within a channel (essentially a channel failure) are revealed during routine checking and testing procedures. Thus, the minimum calibration frequencies set forth are considered acceptable.

Periodic use of the Incore Instrumentation System for power mapping is sufficient to assure that axial and radial power peaks and the peak locations are controlled in accordance with the provisions of the Technical Specifications.

REFERENCE

(1) FSAR, Section 7.1.2.3.4

Table 4.1-1
INSTRUMENT SURVEILLANCE REQUIREMENTS

Channel Description	Check	Test	Calibrate	Remarks
1. Protective Channel Coincidence Logic	NA	MO	NA	
2. Control Rod Drive Trip Breaker	NA	MO	NA	
3. Power Range Amplifier	ES(1)	NA	(1)	(1) Heat balance check each shift. Heat balance calibration whenever indicated core thermal power exceeds neutron power by more than 2 percent.
4. Power Range	ES	MO	MO(1)(2)	(1) Using incore instrumentation. (2) Axial offset upper and lower chambers after each startup if not done previous week.
5. Intermediate Range	ES(1)	PS	NA	(1) When in service.
6. Source Range	ES(1)	PS	NA	(1) When in service.
7. Reactor Coolant Temperature	ES	MO	RF	
8. High Reactor Coolant Pressure	ES	MO	RF	
9. Low Reactor Coolant Pressure	ES	MO	RF	
10. Flux-Reactor Coolant Flow Comparator	ES	MO	RF	
11. Reactor Coolant Pressure Temperature Comparator	ES	MO	RF	

Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
12. Pump-Flux Comparator	ES	MO	RF	
13. High Reactor Building Pressure	DA	MO	RF	
14. High Pressure Injection Logic	NA	MO	NA	
15. High Pressure Injection Analog Channels:				
a. Reactor Coolant Pressure	ES	MO	RF	
b. Reactor Building Pressure (4 psig)	ES	MO	RF	
16. Low Pressure Injection Logic	NA	MO	NA	
17. Low Pressure Injection Analog Channels:				
a. Reactor Coolant Pressure	ES	MO	RF	
b. Reactor Building Pressure (4 psig)	ES	MO	RF	
18. Reactor Building Emergency Cooling and Isolation System Logic	NA	MO	NA	
19. Reactor Building Emergency Cooling and Isolation System Analog Channel Reactor Building Pressure (4 psig)	ES	MO	RF	

Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
20. Reactor Building System Logic	NA	MO	NA	
21. Reactor Building Spray System Analog Channel - Reactor Building High Pressure	NA	MO	RF	
22. Pressurizer Temperature	ES	NA	RF	
23. Control Rod Absolute Position	ES(1)	NA	RF(2)	(1) Check with Relative Position Indicator. (2) Calibrate rod misalignment channel.
24. Control Rod Relative Position	ES(1)	NA	RF(2)	(1) Check with Absolute Position Indicator.
25. Core Flood Tanks				(2) Calibrate rod misalignment channel.
a. Pressure	ES	NA	RF	
b. Level	ES			
26. Pressurizer Level	ES	NA	RF	
27. Letdown Storage Tank Level	DA	NA	RF	
28. Radiation Monitoring Systems	WE(1)	MO	QU	(1) Check functioning of self-checking feature on each detector.
29. High and Low Pressure Injection Systems Flow Channels	NA	NA	RF	

Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
30. Borated Water Storage Tank Level Indicator	WE	NA	RF	
31. Boric Acid Mix Tank;				
a. Level	NA	NA	AN	
b. Temperature	MO	NA	AN	
32. Concentrated Boric Acid Storage Tank:				
a. Level	NA	NA	AN	
b. Temperature	MO	NA	AN	
33. Containment Temperature	NA	NA	RF	
34. Incore Neutron Detectors	MO(1)	NA	NA	(1) Check functioning; including functioning of computer readout or recorder readout.
35. Emergency Plant Radiation Instruments	MO(1)	NA	RF	(1) Battery check.
36. Environmental Monitors	MO(1)	NA	RF	(1) Check functioning.
37. Reactor Manual Trip	NA	PS	NA	
38. Reactor Building Emergency Sump Level	NA	NA	RF	
39. Steam Generator Water Level	WE	NA	RF	
40. Turbine Overspeed Trip	NA	NA	RF	

Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
41. Engineered Safeguards Channel 1 HP Injection Manual Trip	NA	RF	NA	
42. Engineered Safeguards Channel 2 HP Injection Manual Trip	NA	RF	NA	
43. Engineered Safeguards Channel 3 LP Injection Manual Trip	NA	RF	NA	
44. Engineered Safeguards Channel 4 LP Injection Manual Trip	NA	RF	NA	
45. Engineered Safeguards Channel 5 RB Isolation & Cooling Manual Trip	NA	RF	NA	
46. Engineered Safeguards Channel 6 RB Isolation & Cooling Manual Trip	NA	RF	NA	
47. Engineered Safeguards Channel 7 Spray Manual Trip	NA	RF	NA	
48. Engineered Safeguards Channel 8 Spray Manual Trip	NA	RF	NA	

Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks

ES - Each Shift
DA - Daily
WE - Weekly
MO - Monthly

QU - Quarterly
AN - Annually
PS - Prior to startup, if not performed previous week
NA - Not Applicable
RF - Refueling Outage

Table 4.1-2
MINIMUM EQUIPMENT TEST FREQUENCY

	Item	Test	Frequency
1.	Control Rod Movement (1)	Movement of Each Rod	Monthly
2.	Pressurizer Safety Valves	Setpoint	Each Refueling (4)
3.	Main Steam Safety Valves	Setpoint	Each Refueling (4)
4.	Refueling System Interlocks	Functional	Prior to Refueling
5.	Main Steam Stop Valves (1)	Movement of Each Stop Valve	Monthly
6.	Reactor Coolant System Leakage (2)	Evaluate	Daily
7.	Condenser Cooling Water System Gravity Flow Test	Functional	Each Refueling
8.	High Pressure Service Water Pumps and Power Supplies	Functional	Monthly
9.	Spent Fuel Cooling System	Functional	Prior to Refueling
10.	High Pressure and Low Pressure Injection System (3)	Vent Pump Casings	Monthly and Prior to Testing

(1) Applicable only when the reactor is critical.

(2) Applicable only when the reactor coolant is above 200°F and at a steady-state temperature and pressure.

(3) Operating pumps excluded.

(4) Number of safety valves to be tested each refueling shall be in accordance with ASME Codes Section XI, Article IWV-3511, such that each valve is tested at least once every 5 years.

4.4 REACTOR BUILDING

4.4.1 Containment Leakage Tests

Applicability

Applies to containment leakage.

Objective

To verify that leakage from the Reactor Building is maintained within allowable limits.

Specification

4.4.1.1 Integrated Leak Rate Tests

4.4.1.1.1 Design Pressure Leak Rate

The maximum allowable integrated leak rate, La, from the Reactor Building at the 59 psig design pressure, Pp, shall not exceed 0.25 weight percent of the building atmosphere at that pressure per 24 hours.

4.4.1.1.2 Testing at Reduced Pressure

The periodic integrated leak rate test may be performed at a test pressure, P_t, of not less than 29.5 psig provided the resultant leakage rate, Lt, does not exceed a pre-established fraction of La determined as follows:

a. Prior to reactor operation the initial value of the integrated leak rate of the reactor Building shall be measured at design pressure and at the reduced pressure to be used in the periodic integrated leak rate tests. The leak rates thus measured shall be identified as Lpm and Ltm respectively.

b. Lt shall not exceed La(Ltm/Lpm) for values of (Ltm/Lpm) not greater than 0.7.

c. Lt shall not exceed $La(Pt/Pp)^{\frac{1}{2}}$ for values of (Ltm/Lpm) above 0.7.

d. If Ltm/Lpm is less than 0.3, the initial integrated test results shall be subject to review by the NRC to establish an acceptable value of Lt.

4.4.1.1.3 Conduct of Tests

a. The test duration shall be at least 24 hours, except that if both the following conditions are met, the test duration shall be at least 10 hours:

   (1) All test conditions, including the test procedure, shall be similar to the initial integrated leak rate tests.

   (2) When the test is terminated, building pressure shall have stabilized and shall not be increasing.

b. Test accuracy shall be verified by supplementary means, such as measuring the quantity of air required to return to the starting point or by imposing a known leak rate to demonstrate the validity of measurements.

c. Closure of containment isolation valves for the purpose of the test shall be accomplished by the means provided for normal operation of the valves without preliminary exercises or adjustment.

4.4.1.1.4 Frequency of Test

After the initial preoperational leak rate test, two integrated leak rate tests shall be performed at approximately equal intervals between each major shutdown for inservice inspection, to be performed at 10 year intervals. In addition, an integrated leak rate test shall be performed at each 10 year interval, coinciding with the inservice inspection shutdown.

4.4.1.1.5 Conditions for Return to Criticality

a. If Lt is not greater than 50 percent of the value permitted in 4.4.1.1.2, local leak rate testing need not be completed prior to return to criticality following a periodic integrated leak rate test.

b. If Lt is greater than 50 percent and not greater than 100 percent of the value permitted in 4.4.1.1.2, return to criticality will be performed conditioned upon demonstration that locak leakage into the penetration room, measured at full design pressure, accounts for all leakage above 50 percent of that permitted by 4.4.1.1.2. If this cannot be demonstrated within 30 days of returning to criticality, the reactor shall be shut down.

c. If Lt is greater than 100 percent of the value permitted by 4.4.1.1.2, the unit shall not be made critical.

4.4.1.1.6 Corrective Action and Retest

If repairs are necessary to meet the criteria of 4.4.1.1.1 or 4.4.1.1.2, the integrated leak rate test need not be repeated, provided local leak rate measurements are made before and after repair to demonstrate that the leak rate reduction achieved by repairs reduces the overall measured integrated leak rate to an acceptable value.

4.4.1.1.7 Report of Test Results

The results of the initial Containment integrated leak rate test and subsequent periodic tests shall be the subject of a summary technical report which shall be submitted to the Commission within 90 days of completion of the test.

4.4.1.2 Local Leak Rate Tests

4.4.1.2.1 Scope of Testing

The local leak rate shall be measured for each of the following components:

a. Personnel hatch
b. Emergency hatch
c. Equipment hatch seals
d. Fuel transfer tube seals
e. Reactor Building normal sump drain line
f. Reactor coolant pump seal outlet line
g. Reactor coolant pump seal inlet line
h. Quench tank drain line
i. Quench tank return line
j. Quench tank Vent line
k. Normal makeup to Reactor Coolant System
l. High pressure injection line
m. Electrical penetrations
n. Reactor Building purge inlet line
o. Reactor Building purge outlet line
p. Reactor Building sample lines
q. Reactor coolant letdown line

4.4.1.2.2 Conduct of Tests

a. Local leak rate tests shall be performed at a pressure of not less than 59 psig.

b. Acceptable methods of testing are halogen gas detection, soap bubbles, pressure decay, hydrostatic flow or equivalent.

4.4.1.2.3 Acceptance Criteria

The total leakage from all penetrations and isolation valves shall not exceed 0.125 weight percent of the Reactor Building atmosphere per 24 hours.

4.4.1.2.4 Corrective Action and Retest

a. If at any time it is determined that the criterion of 4.4.1.2.3 above is exceeded, repairs shall be initiated immediately.

b. If conformance to the criterion of 4.4.1.2.3 is not demonstrated within 48 hours following detection of excessive locak leakage, the reactor shall be shut down and depressurized until repairs are effected and the local leakage meets the acceptance criterion as demonstrated by retest.

4.4.1.2.5 Test Frequency

Local leak detection tests shall be performed during each reactor shutdown for refueling but in no case at intervals greater than 2 years except that:

a. The equipment hatch and fuel transfer tube seals shall be additionally tested after each opening.

b. The personnel hatch and emergency hatch outer door seals shall be tested at four-month intervals, except when the hatches are not opened during that interval. In no case shall the test interval be longer than 12 months.

4.4.1.3 Isolation Valve Functional Tests

Quarterly, remotely-operated Reactor Building isolation valves shall be stroked to the position required to fulfill their safety function unless such operation is not practical during unit operation. The latter valves shall be tested during each refueling shutdown.

4.4.1.4 Refueling Outage Inspection

A visual examination of the accessible interior and exterior surfaces of the containment structure and its components shall be performed each refueling outage and prior to any integrated leak rate test, to uncover any evidence of deterioration which may affect either the containment's structural integrity or leak-tightness. The discovery of any significant deterioration shall be accompanied by corrective actions in accord with acceptable procedures, non-destructive tests and inspections, and local testing where practical, prior to the conduct of any integrated leak rate test. Results of the inspection shall be reported to the Commission within 90 days of completion.

4.4.1.5 Reactor Building Modifications

Any major modification or replacement of components affecting the Reactor Building integrity shall be followed by either an integrated leak rate test or a local leak rate test, as appropriate, and shall meet the acceptance criteria of 4.4.1.1.4 and 4.4.1.2.3, respectively.

Bases

The Reactor Building is designed for an internal pressure of 59 psig and a steam-air mixture temperature of 286°F. Prior to initial operation, the containment is strength tested at 115 percent of design pressure and leak rate tested at the design pressure. The containment is also leak tested prior to initial operation at approximately 50 percent of the design pressure. These tests verify that the leak rate from Reactor Building pressurization satisfies the relationships given in the specification.

The performance of a periodic integrated leak rate test during unit life provides a current assessment of potential leakage from the containment, in case of an accident that would pressurize the interior of the containment. In order to provide a realistic appraisal of the integrity of the containment under accident conditions, this periodic test is to be performed without preliminary leak detection surveys or leak repairs, and containment isolation valves are to be closed in the normal manner. The test pressure of 29.5 psig for the periodic integrated leak rate test is sufficiently high to provide an accurate measurement of the leak rate and it duplicates the preoperational leak rate test at 29.5 psig. The specification provides a relationship for relating the measured leakage of air at 29.5 psig to the potential leakage at 59 psig. The frequency of the periodic integrated leak rate test is normally keyed to the refueling schedule for the reactor, because these tests can best be performed during refueling shutdowns.

The specified frequency of periodic integrated leak rate tests is based on three major considerations. First is the low probability of leaks in the liner, because of conformance of the complete containment to a 0.25 percent leakage rate at 59 psig during preoperational testing and the absence of any significant stresses in the liner during reactor operation. Second is the more frequent testing, at design pressure, of those portions of the containment envelope that are most likely to develop leaks during reactor operation (penetrations and isolation valves) and the low value (0.125 percent) of leakage that is specified as acceptable from penetrations and isolation valves. Third is the tendon stress surveillance program which provides assurance that an important part of the structural integrity of the containment is maintained.

Leakage to the penetration room, which is permitted to be up to 50 percent of the total allowable containment leakage, is discharged through high efficiency particulate air (HEPA) and charcoal filters to the unit vent. The filters are conservatively said to be 90 percent efficient for iodine removal.

More frequent testing of various penetrations is specified as these locations are more susceptible to leakage than the Reactor Building liner due to the mechanical closure involved. Particular attention is given to testing those penetrations with resilient sealing materials, penetrations that vent directly to the reactor building atmosphere, and penetrations that connect to the Reactor Coolant System pressure boundary. The basis for specifying a maximum leak rate of 0.125 percent from penetrations and isolation valves is that one-half of the actual integrated leak rate is expected from those sources. Valve operability tests are specified to assure proper closure or opening of the Reactor Building isolation valves to provide for isolation of functioning of Engineered Safety Features systems. Valves will be stroked to the position required to fulfill their safety function unless it is established that such testing is not practical during operation. Valves that cannot be full-stroke tested will be part-stroke tested during operation and full-stroke tested during each normal refueling shutdown.

REFERENCES

(1) FSAR, Sections 5 and 13

4.8 MAIN STEAM STOP VALVES

Applicability

Applies to the main steam stop valves.

Objective

To verify the ability of the main steam stop valves to close upon signal and to verify the leak tightness of the main steam stop valves.

Specification

4.8.1 Using Channels A and B, the operation of each of the main steam stop valves shall be tested during each refueling outage to demonstrate a closure time of one second or less in Channel A and a closure time of 15 seconds or less for Channel B.

4.8.2 The leak rate through the main steam stop valves shall not exceed 25 cubic feet per hour at a pressure of 59 psig and shall be tested during each refueling outage.

Bases

The main steam stop valves limit the Reactor Coolant System cooldown rate and resultant reactivity insertion following a main steam line break accident. Their ability to promptly close upon redundant signals will be verified during each refueling outage. Channel A solenoid valves are designed to close all four turbine stop valves in 240 milliseconds. The backup Channel B solenoid valves are designed to close the turbine stop valves in approximately 12 seconds.

Using the maximum 15 second stop valve closing time, the fouled steam generator inventories and the minimum tripped rod worth with the maximum stuck rod worth, an analysis similar to that presented in FSAR Section 14.1.2.9, (but considering a blowdown of both steam generators) shows that the reactor will remain subcritical after reactor trip following a double-ended steam line break.

The main stop valves would become isolation valves in the unlikely event that there should be a rupture of a reactor coolant line concurrent with rupture of the steam generator feedwater header. The allowable leak rate of 25 cubic feet per hour is approximately 25 percent of total allowable containment leakage from all penetrations and isolation valves.

REFERENCES

(1) FSAR Supplement 2, Page 2-7

(2) Technical Specification 4.4.1

ATTACHMENT 2

DUKE POWER COMPANY
OCONEE NUCLEAR STATION

Discussion of Proposed Technical Specification Revisions

Specification 4.0

A new surveillance frequency, "refueling outage," is added with a maximum allowable interval between surveillances of 22 months, 15 days. This change is consistent with the Standard Technical Specifications (STS) for B&W reactors (NUREG 0103, Revision 3). The STS defines the surveillance frequency of "refueling outage" as "at least once per 18 months," with a maximum allowable extension of 25%. This is equivalent to a maximum interval of 22.5 months, to which Duke's proposal is consistent.

Specification 4.1

Surveillance specifications for instruments, components, and systems which require shutdown conditions are changed from a frequency of "annual" to "refueling outage." All changes are consistent with the STS or fall within the requirements of applicable industry codes.

Specification 4.4

The requirement for visual inspection of the containment is changed from "annually" to "each refueling outage." This inspection is not a regulatory requirement. Duke still requires an inspection prior to any integrated leak rate test, which is consistent with 10CFR 50, Appendix J. Duke has reviewed the proposed change and considers it to be acceptable with respect to safe operation of the plant.

Specification 4.8

Closure time and leak rate testing of the main steam stop valves is changed from a frequency of "annually" to "each refueling outage." Since the maximum allowable interval between refueling outage surveillances is less than two years, the new testing frequency for the main steam stop valves is within the time frame specified by 10CFR 50, Appendix J. Duke has reviewed the proposed change and considers it acceptable with respect to safe operation of the plant.