Attachment 3

PROPOSED TECHNICAL SPECIFICATION

4.0 SURVEILLANCE REQUIREMENTS

Specified surveillance intervals may be adjusted plus or minus 25 percent to accommodate normal test and surveillance schedules. Surveillance requirements are not applicable when the plant operating conditions are below those requiring operability of the designated component. However, the required surveillance must be performed prior to reaching the operating conditions requiring operability. For example, instrumentation requiring twice per week surveillance when the reactor is critical need not have the required surveillance when the reactor is shutdown.

Inservice inspection of ASME Code Class 1, 2, 3 components and 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), except where specific written relief has been granted by the NRC pursuant to 10CFR50, Section 50.55a(g)(6)(i).

4.1 OPERATIONAL SAFETY ITEMS

Applicability

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

Objective

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

Specification

a. The minimum frequency and type of surveillance required for reactor protective system and engineered safeguards system instrumentation when the reactor is critical shall be as stated in Table 4.1-1.

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

c. Discrepancies noted during surveillance testing will be corrected and recorded.

d. A power distribution map shall be made to verify the expected power distribution at periodic intervals at least every 10 effective full power days using the incore instrumentation detector system.

Bases

Check

Failures such as blown instrument fuses, defective indicators, faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Futhermore, such failures are, in many cases, revealed by alarm or annunciator action. Comparison 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 plant systems, when the plant is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation.

Calibration

Calibration shall be performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels shall be calibrated at least twice weekly (during steady state operating conditions) against a heat balance standard to compensate for instrumentation drift. During non-steady state operation, the nuclear flux channels shall be calibrated daily to compensate for instrumentation drift and changing rod patterns and core physics parameters.

Table 4.1-2

Minimum Equipment Test Frequency

	Item	Test	Frequency
1.	Control Rods	Rod Drop Times of All Full Length Rods 1/	Each Refueling Shutdown
2.	Control Rod Movement	Movement of Each Rod	Every Two weeks Above Cold Shutdown Conditions
3.	Pressurizer Code Safety Valves	Per Specification 4.0	Per Specification 4.0
4.	Main Steam Safety Valves	Per Specification 4.0	Per Specification 4.0
5.	Refueling System Interlocks	Functioning	Start of Each Refueling Shutdown
6.	Reactor Coolant System Leakage	Evaluate	Daily
7.	Deleted		
8.	Reactor Building Isolation Trip	Functioning	Every 18 Months
9.	Service Water Systems	Per Specification 4.0	Per Specification 4.0
10.	Spent Fuel Cooling System	Functioning	Per Specification 4.0
11.	Decay Heat Removal System Isolation Valves	Functioning	Per Specification 4.0
12.	Flow Limiting Annulus on Main Feedwater Line at Reactor Building Penetration	Verify, at normal operating conditions, that a gap of at least 0.025 inches exists between the pipe and the annulus	One year, two years, three years, and every five years thereafter measured from date of initial test

1/ Same as tests listed in Section 4.7

Table 4.1-2 (Continued)
Minimum Equipment Test Frequency

Item	Test	Frequency
13. SLBIC Pressure Sensors	Calibrate	Every 18 Months
14. Main Steam Isolation Valves	Per Specification 4.0	Per Specification 4.0
15. Main Feedwater Isolation Valves	Per Specification 4.0	Per Specification 4.0
16. Reactor Internals Vent Valves	Demonstrate Operability By: a. Conducting a remote visual inspection of visually accessible surfaces of the valve body and disc sealing faces & evaluating any observed surface irregularities, b. Verifying that the valve is not stuck in an open position, and c. Verifying through manual actuation that the valve is fully open with a force of ≤ 400 lbs. (applied vertically upward).	Each refueling Shutdown

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 Prior to initial unit operation, an ultrasonic test survey shall be made of reactor coolant system pressure boundary welds as required to establish preoperational integrity and baseline data for future inspections.

4.2.2 Post-operational inspections of components shall be made in accordance with the methods and intervals indicated in Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10CFR50, Section 50.55a(g), except where specific written relief has been granted by the NRC.

4.2.3 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.

4.2.4 Complete surface and volumetric examination of the reactor coolant pump flywheels will be conducted coincident with refueling or maintenance shutdowns such that within a 10 year period after start-up all four reactor coolant pump flywheels will be examined.

4.2.5 The reactor vessel material irradiation surveillance specimens removed from the reactor vessel in 1976 shall be installed, irradiated in and withdrawn from the Davis-Besse Unit No. 1 reactor vessel in accordance with the schedule shown in Table 4.2-1. Following withdrawal of each capsule listed in Table 4.2-1, Arkansas Power & Light Company shall be responsible for testing the specimens and submitting a report of test results in accordance with 10CFR50, Appendix H.

TABLE 4.2-1

ANO-1 CAPSULE ASSEMBLY WITHDRAWAL SCHEDULE AT DAVIS-BESSE 1

CAPSULE	INSERTION/WITHDRAWAL
ANI-E	Has been withdrawn for testing
ANI-B	Withdraw following 1st cycle at Davis-Besse 1
ANI-A	Withdraw following 3rd cycle at Davis-Besse 1
ANI-C	Withdraw following 7th Cycle at Davis-Besse 1
ANI-D	Insert in location WZ (upper) prior to 4th cycle at Davis-Besse 1; withdraw following 12th cycle
ANI-F	Insert in location YZ (upper) prior to 4th cycle at Davis-Besse 1; withdraw following 11th cycle

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 50.55a, to the extent practicable within limitations of design, geometry and materials of construction.

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 4.2.8, the definition of Regulatory Guide 1.16, Revision 4 (August 1975) applies for the term "commercial operation". 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)).

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 B 31.7, and ASME Boiler and Pressure Vessel Code, Section XI.

For normal opening, the integrity of the Reactor Coolant System, in terms of strength, is unchanged. If the system does not leak at 2285 psig (operating pressure +100 psi; ±50 psi is normal system pressure fluctuation), it will be leak tight during normal operation.(1)

REFERENCES

FSAR, Section 4

4.4.1.2.5 Test Frequency

Local leak detection tests shall be performed during each reactor shutdown for refueling or other convenient intervals, but in no case at intervals >2 years except that:

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

(b) If a personnel hatch or emergency hatch door is opened when reactor building integrity is required, the affected door seal shall be tested. In addition, a pressure test shall be performed on the personnel and emergency hatches every six months.

4.4.1.3 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 test, as appropriate, and shall meet the acceptance criteria specified in 4.4.1.1 and 4.4.1.2 respectively.

4.4.1.4 Isolation Valve Functional Tests

No additional Surveillance Requirements other than those required by Specification 4.0 shall be required.

4.4.1.5 Visual Inspection

A visual examination of the accessible interior and exterior surfaces of the reactor building structure and its components shall be performed during each refueling shutdown and prior to any integrated leak test, to uncover any evidence of deterioration which may affect either the reactor building's structural integrity or leak-tightness. The discovery of any significant deterioration shall be accompanied by corrective actions in accord with acceptable procedures, nondestructive tests, and inspections, and local testing where practical prior to the conduct of any integrated leak test. Such repairs shall be reported as part of the test results.

Bases (1)

The reactor building is designed for an internal pressure of 59 psig and a steam-air mixture temperature of 285F. Prior to initial operation, the reactor building will be strength tested at 115% of design pressure and leak rate tested at the design pressure. The reactor building will also be leak tested prior to initial operation at not less than 50% of

4.5.1.1.3 Core Flooding System

No additional Surveillance Requirements other than those required by Specification 4.0 shall be required.

4.5.1.2 Component Tests

4.5.1.2.1 Pumps

No additional Surveillance Requirements other than those required by Specification 4.0 shall be required.

4.5.1.2.2 Valves - Power Operated

No additional Surveillance Requirements other than those required by Specification 4.0 shall be required.

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 will be rotated to another high pressure injection pump. This will help verify that the high pressure injection pumps are operable.

The requirements of the 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 will verify 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 borated water storage tank recirc line. This allows water to be pumped from the borated water storage tank through each of the injection lines and back to the tank.

REFERENCE

FSAR Section 6

(b) The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly.

4.5.2.2 Component Tests

4.5.2.2.1 Pumps

No additional Surveillance Requirements other than those required by Specification 4.0 shall be required.

4.5.2.2.2 Valves

No additional Surveillance Requirements other than those required by Specification 4.0 shall be required.

Bases

The reactor building cooling system and reactor building spray system are designed to remove the heat in the reactor building atmosphere to prevent the building pressure from exceeding the design pressure.

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 at 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 smoke 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.

4.8 EMERGENCY FEEDWATER SYSTEM

Applicability

Applies to the periodic testing of the turbine and electric motor driven emergency feedwater pumps.

Objective

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

Specification

4.8.1 Test

1. The turbine and electric motor driven emergency feedwater pumps shall be operated every three months for a minimum of one hour.

2. The emergency feedwater valves shall be cycled every three months.

3. Once every 18 months, a functional test of the emergency feedwater system shall be made using the electric motor driven emergency feedwater pump.

4.8.2 Acceptance Criteria

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

Bases

The three (3) month testing frequency will be sufficient to verify that both emergency feedwater pumps are operable. Verification of correct operation will be made both from the control room instrumentation and direct visual observation of the pumps. The cycling of the emergency valves will be done coincident with the pump testing, but not concurrently so that cold emergency feedwater is not pumped to the steam generator.

The functional test, performed once every 18 months, will verify that the flow path to the steam generators is open and that water reaches the steam generators from the emergency feedwater system. The test is done during shutdown to avoid thermal cycle to the emergency feedwater nozzles on the steam generator due to the lower temperature of the emergency feedwater.