JAMES A. FITZPATRICK - 24 MONTH OPERATING CYCLE

CONTAINMENT SYSTEMS
SURVEILLANCE TEST EXTENSIONS

JAF-RPT-MULTI-01116 Rev. 1

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EXECUTIVE SUMMARY

The Fitzpatrick plant will be operating on a 24 month fuel cycle. This longer cycle length has a direct effect on surveillance testing activities that are currently performed on a 18 month, operating cycle or refuel outage basis.

At Fitzpatrick, the Primary and Secondary Containment Systems are routinely inspected, tested, and maintained to provide high reliability. These systems are subject to tests which verify the operability of several subsystems, such as: the drywell, the pressure suppression pool or torus, the connecting vent system, the vacuum relief system, the standby gas treatment system, and the reactor building isolation and control system.

Test frequencies are mandated by the plants technical specifications, operational requirements, and inservice inspection schedules.

This study evaluates the changes to surveillance requirements to support a nominal twenty four month fuel cycle. Justification is provided, where appropriate, to support test extensions.

Our evaluations assure that the current surveillance test intervals can be extended to support a nominal 24 month operating cycle. (See Attachment A)

JAF 24 MONTH OPERATING CYCLE
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I. PURPOSE

The Fitzpatrick plant will be operating on a 24 month fuel cycle. To avoid either an 18 month surveillance outage or an mid-cycle outage, changes are required to the Containment Systems surveillance test intervals prescribed by the Fitzpatrick Technical Specifications. Substantiating the impacts of the longer cycle length on the Containment Systems surveillance test activities requires a comprehensive review of the system, its individual components, and the integrated effect of test and maintenance activities on operability.

II. EVALUATION METHODOLOGY

The once per cycle surveillance record for the applicable system components will be evaluated for a time period defined in the evaluation section.

System instruments that will be having their calibration frequency extended to 24 months will have a drift evaluation developed. The drift evaluation will justify the calibration frequency extension or the calibration frequency will not change.

System components that will be having their surveillance frequency extended to 24 months will be evaluated in the following manner. If surveillance records reveals more than one failure to meet acceptance criteria per component, then the surveillance frequency can not be extended. One failure per component is viewed as acceptable. If exceptions are to be taken, they will be presented in the evaluation section.

III. SYSTEM SAFETY FUNCTIONS

The Containment Systems of the James A. Fitzpatrick Nuclear Power Plant are part of a "multibarrier" concept. The primary barrier is the Primary Containment, which is of the pressure suppression type. The second barrier is the Secondary Containment which is provided to minimize the ground level release of airborne radioactive materials. The fuel, fuel cladding, and the Reactor Coolant Pressure Boundary form additional barriers to the release of fission products.

The Primary Containment is of the pressure suppression type and consists of the drywell, the pressure suppression pool or torus, the Connecting Vent System between the drywell and the torus, isolation valves, the Vacuum Relief System, and the RHR subsystems for containment cooling. The drywell is a steel pressure vessel in the shape of a light bulb, and the pressure suppression chamber is a torus-shaped steel pressure vessel located below and encircling the drywell. The Primary Containment is designed to withstand the pressure and temperatures resulting from a breach of the Reactor Coolant Pressure Boundary up to and including an instantaneous circumferential break of the reactor recirculation piping and to provide a holdup for decay of any radioactive material to be released.

In the event of a Reactor Coolant Pressure Boundary failure within the drywell, reactor water and steam are released in to the drywell gas space. The resulting increased drywell pressure forces a mixture of non-condensable gases, steam, and water through the connecting vents into the pressure suppression pool. The steam condenses rapidly in the water filled pressure suppression pool resulting in a rapid pressure reduction in the drywell. Non-condensable gases transferred during reactor blowdown to the pressure suppression chamber pressurize the chamber and are subsequently vented back to the drywell through the Vacuum Relief System as the pressure in the drywell drops below that in the pressure suppression chamber.

The Secondary Containment encloses the Primary Containment, the refueling and reactor servicing areas, the new and spent fuel storage facilities, and other reactor auxiliary systems. The Secondary Containment serves as the containment during reactor refueling and maintenance operations when the primary containment is open, and as an additional barrier when the Primary Containment is functional. The Secondary Containment consists of the Reactor Building, the Standby Gas Treatment System, the main stack, and the Reactor Building Isolation and Control System.

The Secondary Containment utilizes four different features to mitigate the consequences of the postulated loss of coolant accident inside the drywell and the refueling accident inside the secondary containment. The first feature is a negative pressure barrier which prevents the ground level release of fission products by exfiltration. The second feature is a large low-leakage secondary containment volume which provides a holdup time for fission product decay prior to release. The third feature is the removal of particulates and iodides by filtration prior to release. The fourth feature is the exhausting of the secondary containment atmosphere through an elevated release point which aids in dispersion of the effluent by atmospheric dispersion. Each of these features is provided by a different combination of subsystems: the first by the Reactor Building, the Reactor Building Isolation and Control System, and the Standby Gas Treatment exhaust fans; the second by the Reactor Building and the Reactor Building Isolation and Control System, the third by the Standby Gas Treatment System filters; and the fourth by the main stack.

The Secondary Containment is designed to withstand the design basis earthquake and to provide holdup, treatment, and an elevated release point for any fission products released to it. In addition, the Reactor Building is designed to protect the Emergency Core Cooling System and auxiliary systems located in the building from all postulated environmental events.

IV. SURVEILLANCE TEST EVALUATIONS

The operability of systems and components required by the plant's safety analyses is established by the surveillance requirements contained in the Technical Specifications. Surveillance testing, by definition, can only identify that a component or a system is incapable of performing its safety function (i.e., inoperable).

Containment Systems surveillance testing and inspection activities were thoroughly evaluated to determine the impacts of a 24 month operating cycle. The longer cycle length requires an extension of the following tests.

1.	ST-7A	Standby Gas Treatment Manual Bypass Operation, Heater Capacity, Filter Differential Pressure, and Downstream Piping Leak Tests
2.	ST-15B	Suppression Chamber and Drywell Deterioration Inspection
3.	ST-39B	Type B and C LLRT of Containment Penetrations
4.	ST-39D	Reactor Building Leak Rate Test
5.	ST-39E	Pressure Suppression Chamber - Drywell Vacuum Breaker Leak Test (IST)
6.	ST-39F	Primary Containment Integrated Leakage Rate (Type A) Test
7.	ST-39J	Leak Testing of RHR and Core Spray Testable Check Valves (IST)
8.	(Multi)	Primary Containment Isolation Valve Simulated Automatic Initiation and Closure Timing

A. Surveillance Test Changes

The decision to extend surveillance test intervals considers:

(1) the function of the test in determining overall system operability, and

(2) the integrated effect of testing and maintenance activities on system operability.

Also considered, but normally not utilized as a primary justification for surveillance test extension is the burden of testing at power; i.e., testing that could lead to a plant transient, testing that results in unnecessary equipment wear, and/or testing that leads to radiation exposure of plant personnel.

1. Standby Gas Treatment Operation, Heater Capacity, Filter Differential Pressure, and Downstream Piping Leak Tests (ST-7A)

The Standby Gas Treatment System (SBGT) ensures that two design features of the Secondary Containment are met. The system provides secondary containment with a negative pressure to control leakage, and also filtration for removal of particulates and iodides prior to release through the stack. The SBGT includes two identical filter trains, each consisting of the following components: Demister, 39 kW Electric Air Heater, Prefilter, High Efficiency Particulate Absolute Prefilter, Activated Charcoal Filter, High Efficiency Particulate Absolute Afterfilter, and Fan. Each train is a full capacity system that has the ability to maintain a -0.25 inches water gauge differential pressure in the reactor building relative to the atmosphere.

The purpose of this surveillance is to verify the operability of the manual fan suction cross-tie bypass valves for filter cooling (01-125SGT-3A & B), operate each SBGT train to determine filter differential pressure, measure heater power, and inspect for leakage at pressurized piping joints. ST-7A was completed on 5/85, 6/85, 1/87, 10/88 (B side only), 11/88, 6/89, 5/90, 4/91, 3/92, 6/92, 9/92, 12/93 and 5/95 and there were no component failures identified except on 5/95 a gasket on the fan discharge piping had to be replaced.

The SBGT Automatic Initiation test is performed in ST-34B, Reactor Building Exhaust Radiation Monitors Instrumentation/Logic Functional and Simulated Automatic Actuation. The test frequency for ST-34B is once per quarter which is adequate to detect equipment deterioration when evaluating for once per operating cycle testing. ST-34B testing of the SBGT Automatic Initiation adequately supports a 24 month operating cycle.

2. Suppression Chamber and Drywell Deterioration Inspection (ST-15B)

The purpose of this procedure is to inspect the interior surfaces of the drywell and above the water line of the pressure suppression chamber (torus) for signs of deterioration. This is accomplished by examining the surfaces for signs of deterioration such as scaling, rusting, and/or paint chipping (Reference 11).

All coatings used inside primary containment are qualified for JAF specific normal and design basis accident environmental conditions in accordance with ANSI N5.12-1974 and ANSI N101.2-1972.

The coating system selection and application follows the guidelines of Regulatory Guide 1.54 "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants", which references ANSI N101.4-1972 as an acceptable plan for compliance with NRC requirements for adequate quality assurance for coatings used in nuclear containments. In lieu of the inspection defined in section 6.2.4 of ANSI N101.4-1972, inspection is in accordance with ANSI N5.12-1974 section 10, Inspection for Shop and Field work. The reference substitution of ANSI N5.12 as the basis for inspection, rather than ANSI N5.9 reflects a revision to a standard referenced in the based document, ANSI N101.4-1972.

ST-15B was completed on 5/85, 4/87 (torus only), 11/88, 6/90, 4/92, 12/92, 4/94 and 1/95 and there were no failures identified, degradation of the drywell and torus has been minimal.

3. Type B and C Local Leak Rate Test (LLRT) of Containment Penetrations (ST-39B)

The purpose of this surveillance test is to assess potential leakage from the containment penetrations and isolation valves based on the calculated peak containment internal pressure following an accident. Local leak rate testing is currently performed once per refueling outage and requires evaluation for extension with the 24 month operating cycle.

Technical Specifications and 10 CFR 50 Appendix J require that Type B and C leakage rate testing of containment penetrations be performed during each reactor shutdown for refueling, but in no case greater than 2 years (Reference 15). There is no grace period allowed on this testing interval (i.e. no +25%). Therefore in order to process a technical specification change to support a 24 month cycle (i.e. 24 months +25%), an exemption to Appendix J requirements would be needed.

Generic Letter 91-04 anticipated that licensees would desire an extension of the current testing interval to support 24 month cycle extensions. The Generic Letter provided guidance which detailed the information needed to support the required Appendix J exemption. Requesting the exemption as outlined in the Generic Letter would result in the lowering of JAFs already conservative combined leakage limit of 0.6La to 0.5La.

Since the Generic Letter was issued, the NRC has taken action to implement a new Appendix J ruling which would allow performance based containment testing, rather than the prescriptive test requirements currently contained in the regulation. This new ruling would support a 24 month cycle because it would allow the 25% grace period to be applied to the Type B and C testing intervals. Final ruling is expected in October 1995.

To avoid having to reduce the combined leakage limit from 0.6La to 0.5La, and to avoid making 2 changes to the same technical specification pages, JAF Licensing and Technical Services departments have decided not to use the Generic Letter 91-04 approach for the Appendix J exemption. Once

In the event that the new Appendix J ruling does not become finalized in sufficient time for the 1996 refueling outage, the following options are available:

1. Shutdown prior to November 1996 and test the valves within the required test frequency.

2. Pursue the Appendix J exemption as outlined in GL 91-04.

3. Pursue a one time schedular exemption if the start of the 1996 refueling outage is delayed past November 1996.

4. Reactor Building Leak Rate Test (ST-39D)

The purpose of this test is to demonstrate secondary containment leak rate integrity using the Standby Gas Treatment System. The surveillance consists of running the SBGT at less than or equal to 6000 scfm flow with the following conditions: 1) Reactor Building and the Reactor Building Ventilation System isolated, 2) one door open in each airlock, and 3) the track bay door seals deflated to simulate loss of instrument air. During the surveillance, each train of the SBGT is operated separately (at less than 6000 scfm) to maintain reactor building differential pressure more negative than -0.25 inches of water (Reference 22).

ST-39D was completed on 2/85, 8/88, 3/90, 10/90, 1/91, 1/92, 4/92, 5/93, 6/93, 11/94 and 12/94 and there were no component failures identified except on 1/92 when high winds outside the plant caused some problems. The test passed without exception on 4/92.

5. Pressure Suppression Chamber - Drywell Vacuum Breaker Leak Test - IST (ST-39E)

Vacuum breaker valves are utilized to prevent the primary containment pressure from dropping below its vacuum rating relative to the external design pressure by drawing non-condensable gases back to the drywell from the pressure suppression chamber. The vacuum breakers are 30 inch diameter swing check valves equipped with counterweighted arms ensuring the valves remain seated until torus pressure exceeds pressure in the drywell by 0.5 psid. They are normally sealed with increased pressure on the drywell side.

During a postulated design basis accident, steam is condensed by being blown down through water in the pressure suppression chamber. A certain amount of bypass leakage (bypassing the torus water) is expected through the vacuum breakers. This test is conducted to determine the total equivalent bypass area leakage. As part of the routine surveillance, a leak test is conducted at each refueling outage to ensure the integrity of the drywell - suppression chamber leakage path. The drywell to suppression chamber bypass leakage is limited to 71 scfm of nitrogen (Reference 24).

ST-39E was completed on 2/85, 1/87, 8/88, 3/90, 7/91, 8/91, 12/92, 4/94 and 11/94 and there were no failures identified except on 7/91 when 39SAS-40 would not close and the sensors were not tracking. The test passed without exception on 8/91.

The Vacuum Breaker visual inspection and force test is performed in ST-15F, Torus to Drywell Vacuum Breaker setpoint test. The test frequency for ST-15F is once per quarter which is adequate to detect equipment deterioration when evaluating for once per operating cycle testing. ST-15F testing of the vacuum breaker force test and visual inspection adequately supports a 24 month operating cycle.

6. Primary Containment Integrated Leakage Rate (Type A) Test - (ST-39F)

The purpose of this test is to demonstrate primary containment integrity by meeting Technical Specification leakage limits.

The Type A test measures the primary reactor containment overall leakage rate. The containment is pressurized to the test pressure of 45 psig using temporary air compressors. Containment pressure and temperature are allowed to stabilize for a minimum of 4 hours. After the Containment has satisfied the stabilization criteria, the Type A test period begins. Containment pressure, temperature, and vapor pressure are monitored during the test period. The appropriate data is input frequently into the PCILRT computer program to calculate the primary containment integrated leakage rate. A verification test using a superimposed leakage is then used to verify the Type A test results.

ST-39F was completed on 5/85, 4/87, 6/90 and 3/95 and there were no failures identified.

7. Leak Testing of RHR and Core Spray Testable Check Valves - IST (ST-39J)

The purpose of this test is to demonstrate that valves 10AOV-68A, 10AOV-68B, 14AOV-13A and 14AOV-13B have leakage rates which are within acceptable limits. The procedure subsections for each valve can be performed in any order as necessary to facilitate plant operations or maintenance.

ST-39J was completed on 2/85 (10AOV-68B only), 1/87, 3/87 (10AOV-68A only), 8/88 (10AOV-68A & 14AOV-13A only), 9/88 (14AOV-13B only), 10/88 (10AOV-68B only), 4/90, 2/92, 4/92, 10/93 (10AOV-68B & 14AOV-13B only), 11/93 (10AOV-68A & 14AOV-13A only), 12/94 (10AOV-68B & 14AOV-13B only) and 1/95 (10AOV-68A & 14AOV-13A only) and there were no component failures identified except on 2/92 when 14AOV-13A tested unsatisfactory. The test passed without exception on 4/92.

8. Primary Containment Isolation Valve Simulated Automatic Initiation and Closure Timing (Multi).

The purpose of these tests are to simulate automatic initiation and verify acceptable closure timings of the power operated primary containment isolation valves.

ST-1B, ST-1S, ST-2AL, ST-2AM, ST-3P, ST-4N, ST-24J, ST-26M and ST-34B have a test frequency of once per quarter which is adequate to detect equipment deterioration when the testing requirement is evaluating for once per operating cycle. These tests adequately support a 24 month operating cycle.

ST-2G, ST-4F, ST-24D, ST-26I and ST-34A have a test frequency of semi-annual which is adequate to detect equipment deterioration when the testing requirement is evaluating for once per operating cycle. These tests adequately support a 24 month operating cycle.

ST-4P has a test frequency of annually which means it will be performed at least twice during a 24 month operating cycle which is adequate to detect equipment deterioration when the testing requirement is evaluating for once per operating cycle. This test adequately supports a 24 month operating cycle.

ST-2P is evaluated in report JAF-RPT-MULTI-01561 R/1, Surveillance Test Extensions for ECCS Mechanical Systems.

V. SUMMARY AND CONCLUSIONS

To support the 24 month fuel cycle, changes are proposed to the containment surveillance test intervals for the following system functional tests (See Attachment A).

1.	ST-7A	Standby Gas Treatment Manual Bypass Operation, Heater Capacity, Filter Differential Pressure, and Downstream Piping Leak Tests.
2.	ST-15B	Suppression Chamber and Drywell Deterioration Inspection.
3.	ST-39D	Reactor Building Leak Rate Test.
4.	ST-39E	Pressure Suppression Chamber - Drywell Vacuum Breaker Leak Test (IST)
5.	ST-39J	Leak Testing of RHR and Core Spray Testable Check Valves (IST)

Technical Specification changes will be required by the extension of certain tests. The proposed changes to the Fitzpatrick Technical Specifications are given in Attachment B.

VI. REFERENCES

1. US Nuclear Regulatory Commission Generic Letter 91-04, "Changes in Technical Specifications Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," April 2, 1991.

2. Torus to Drywell Vacuum Breaker Setpoint test (ST-15F).

3. MSIV Fast Close & MSL Drain Isolation Valve Test (ST-1B).

4. SBGT Manual Bypass Valve Operation, Heater Capacity, Filter dP, and Downstream Piping Leak Tests (ST-7A).

5. Shutdown Cooling Containment Isolation Valve Test (ST-1S).

6. Standby Gas Treatment System, OP-20.

7. RHR Isolation Valve Control Logic Functional (ST-2G).

8. Report JAF-RPT-MULTI-01561 R/1, Surveillance Test Extensions For ECCS Mechanical Systems.

9. USAEC, Oak Ridge National Laboratory, "Aging, Weathering, and Poisoning of Impregnated Charcoals Used for Trapping Radioiodine," March 1970.

10. Core Spray Flow Rate and Valve Test (ST-3P).

11. Suppression Chamber and Drywell Deterioration Inspection (ST-15B).

12. FitzPatrick Final Safety Analysis Report.

13. Fitzpatrick Technical Specifications.

14. Code of Federal Regulations, 10 CFR 50, Appendix J.

15. Type B and C LLRT of Containment Penetrations (ST-39B).

16. HPCI Auto Isolation Logic Functional and Simulated Auto Actuation Test (ST-4F).

17. NYPA Containment Isolation Valve Replacement Report, October, 1990.

18. BWROG Plant Survey on Leak Rate Testing, August 1987.

19. NYPA Letter to the NRC, JPN-88-012, April 8, 1988.

20. Failure Prevention, Inc., analysis of Selected Containment Isolation Valve Failures and a Review of Local Leak Rate Testing Programmatic Issues at the James A FitzPatrick Nuclear Power Plant, July 1992.

21. USNRC Letter to R.E. Beedle, "Issuance of and Exemption to the Requirements of 10 CFR Part 50, Appendix J, Paragraph III.D.3 - Indian Point Nuclear Generating Unit No. 3 (TAC No. 1784137)" Feb. 19, 1993.

22. Reactor Building Leak Rate Test (ST-39D).

23. Reactor Building Ventilation and Cooling (OP-51A)

24. Pressure Suppression chamber - Drywell Vacuum Breaker Leak Test (ST-39E).

25. HPCI Flow Rate and IST (ST-4N)

26. Primary Containment Integrated Leakage Rate (Type A) Test (ST-39F)

27. Leak Testing of RHR and Core Spray Testable Check Valves-IST (ST-39J)

28. Reactor Building Exhaust Radiation Monitors Instrumentation/Logic Functional and Simulated Automatic Actuation (ST-34B)

29. HPCI Quick-start Transient Monitoring Test (ST-4P)

30. RCIC Auto Isolation Logic Functional and Simulated Auto Actuation Test (ST-24D)

31. RCIC Flow rate and IST (ST-24J)

32. RWCU Isolation Logic Functional and Simulated Auto Actuation Test (ST-26I)

33. RWCU Valve Testing (ST-26M)

34. PCIS Group 2 Logic Functional and Simulated Auto Actuation Test (ST-34A)

ATTACHMENT A

CONTAINMENT SURVEILLANCE TEST CHANGES

PROCEDURE	TS SECTION	CHANGE
ST-7A (ST-34B)	4.7.B.1	18M To 24M
ST-15B	4.7.A.1	18M to 24M
ST-39B	4.7.A.2.e	No Change
ST-39D	4.7.C.1.c	18M to 24M
ST-39E (ST-15F)	4.7.A.5	18M to 24M
ST-39F	4.7.A.2.a	No Change
ST-39J	4.7.A.2.e	18M to 24M
ST-1B&1S ST-2G, 2AL&2AM ST-3P ST-4F, 4N&4P ST-24D&24J ST-26I&26M ST-34A&34B	4.7.D.1.a	No Change

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ATTACHMENT B

TECHNICAL SPECIFICATIONS CHANGES

4.7 (cont'd)

4.7.A.2

Type A test shall be performed at each plant shutdown for refueling or approximately every ~~18~~ 24 months, whichever occurs first, until two consecutive Type A tests meet the acceptance criteria.

b. Type B tests (Local leak rate testing of containment penetrations)

(1.) All preoperational and periodic Type B tests shall be performed by local pneumatic pressurization of the containment penetrations, either individually or in groups, at a pressure not less than Pa, and the gas flow to maintain Pa shall be measured.

(2.) Acceptance criteria

The combined leakage rate of all penetrations and valves subject to Type B and C tests shall be less than 0.60 La, with the exception of the valves sealed with fluid from a seal system.

4.7 (cont'd)

(5) Type C test.

Type C tests shall be performed during each reactor shutdown for refueling but in no case at intervals greater than two years.**

(6) Other leak rate tests specified in Section 4.7.d shall be performed during ~~each reactor~~ shutdown for refueling but in no case at intervals greater than ~~two years~~.

30 months

f, Containment modification*

Any major modification, replacement of a component which is part of the primary reactor containment boundary, or resealing a seal-welded door, performed after the preoperational leakage rate test shall be followed by either a Type A, Type B, or Type C test, as applicable, for the area affected by the modification. The measured leakage from this test shall be included in the test report. The acceptance criteria as appropriate, shall be met. Minor modifications, replacements, or resealing of seal-welded doors, performed directly prior to the conduct of a scheduled Type A test do not require a separate test.

* In accordance with an exemption from 10 CFR 50 Appendix J, a Type A, B, or C test is not required for the replacement of piping and welds which constitute the Core Spray System minimum flow lines (3"-W23-152-7A, B) during the 1993 maintenance outage.

** In accordance with an exemption from 10 CFR 50 Appendix J, the Type C test of the shutdown cooling isolation valves (10MOV-17 and 10MOV-18) may be deferred until refueling outage Reload 11/ Cycle 12.