TABLE OF CONTENTS

8404050197 840402
PDR ADOCK 05000250
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TABLE OF CONTENTS (Continued)

Amendment Nos. ____ and ____

TABLE OF CONTENTS (Continued)

96 90

TABLE OF CONTENTS (Continued)

Amendment Nos. ____ and ____

LIST OF TABLES

LIST OF FIGURES

3.3 CONTAINMENT

Applicability: Applies to the integrity of the containment.

Objective: To define the operating status of the containment.

Specification: 1. CONTAINMENT INTEGRITY

a. The containment integrity (as defined in 1.5) shall not be violated unless the reactor is in the cold shutdown condition. Specification 3.0.1 applies to 3.3.1.

b. The containment integrity shall not be violated when the reactor vessel head is removed unless the reactor is in the refueling shutdown condition.

2. INTERNAL PRESSURE

If the internal pressure exceeds 3 psig or the internal vacuum exceeds 2 psig, the condition shall be corrected within 8 hours or the reactor shall be brought to hot shutdown.

3. CONTAINMENT ISOLATION VALVES

With $K_{eff} \geq 0.99$. % thermal power excluding decay heat ≥ 0, and an average coolant temperature $T_{avg} \geq 200$ F, the following conditions shall be met:

The containment isolation valves shall be operable in accordance with the requirements of Specification 4.0.3., or the valve is closed.

5. TWO residual heat removal pumps shall be operable.

6. TWO residual heat exchangers shall be operable.

7. All valves, interlocks and piping associated with the above components and required for post accident operation, shall be operable except valves that are positioned and locked. Valves 864-A, B, 862-A, B, 865-A, B, C; 866-A, B shall have power removed from their motor operators by locking open the circuit breakers at the Motor Control Centers. The air supply to valve 758 shall be shut off to the valve operator.

b. During power operation, the requirements of 3.4.1a may be modified to allow one of the following components to be inoperable (including associated valves and piping) at any one time except for the cases stated in 3.4.1.b.2. If the system is not restored to meet the requirements of 3.4.1a within the time period specified, the reactor shall be placed in the hot shutdown condition. If the requirements of 3.4.1a are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition. Specification 3.0.1 applies to 3.4.1.b.

1. ONE accumulator may be out of service for a period of up to 4 hours.

2. ONE of FOUR safety injection pumps may be out of service for 30 days. A second safety injection pump may be out of service, provided the pump is restored to operable status within 24 hours. Compliance with the requirements of Specification 4.0.3 shall be verified for the remaining 2 pumps.

3. ONE channel of heat tracing on the flow path may be out of service for 24 hours.*

4. ONE residual heat removal pump may be out of service, provided the pump is restored to operable status within 24 hours. Compliance with the requirements of Specification 4.0.3 shall be verified for the other pump.

add *

add

* See reference (11) on page B.3.4-2

5. ONE residual heat exchanger may be out of service for a period of 24 hours. Compliance with the requirements of Specification 4.0.3 shall be verified for the remaining heat exchanger.

6. Any valve in the system may be inoperable provided repairs are completed within 24 hours. Compliance with the requirements of Specification 4.0.3 shall be verified for all valves that provide the duplicate function.

7. To permit temporary operation of the valve, e.g., for surveillance of valve operability, for the purpose of valve maintenance, etc., the valves specified in 3.4.1.a.7 may be unlocked and may have supplied air or electric power restored for a period not to exceed 24 hours.

c. During power operation three Reactor Coolant Loops shall be in operation.

1. With less than three Reactor Coolant Loops in operation, the reactor must be in hot shutdown within one hour.

d. In hot shutdown at least two Reactor Coolant Loops shall be operable and at least one Reactor Coolant Loop shall be in operation.*

1. With less than two Reactor Coolant Loops operable, restore the required Coolant Loops to operable status within 72 hours or reduce T_{avg} to less than or equal to 350 F within the next 12 hours.

2. With no Reactor Coolant Loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required Coolant Loop to operation.

e. With average coolant temperature less than 350 F, at least two Coolant Loops shall be operable or immediate corrective action must be taken to return two Coolant Loops to operable as soon as possible. One of these Coolant Loops shall be in operation.*

1. With no Coolant Loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required Coolant Loop to operation.

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* All reactor coolant pumps and residual heat removal pumps may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and 2) core outlet temperature is maintained as last 10 F below saturation temperature.

f. In refueling shutdown, at least one residual heat removal Coolant Loop shall be in operation or all operations involving an increase in the reactor decay heat load or a reduction in boron concentration in the Reactor Coolant System must be suspended, and all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed in four hours. As an exception, the single residual heat removal Coolant Loop may be removed from operation during the performance of core alterations in the vicinity of the reactor pressure vessel hot legs, provided core outlet temperature is maintained below 160 F.

g. In refueling shutdown, when the water level above the top of the pressure vessel flange is less than 23 feet, two residual heat removal Coolant Loops shall be operable or action to return two residual heat removal Coolant Loops to operable shall be taken as soon as possible.

2. EMERGENCY CONTAINMENT COOLING SYSTEMS

a. The reactor shall not be made critical, except for low power physics tests unless the following conditions are met:

1. Three emergency containment cooling units are operable.

2. Two containment spray pumps are operable.

3. All valves and piping associated with the above components, and required for post accident operation, are operable.

b. During power operation, the requirements of 3.4.2a may be modified to allow one of the following components to be inoperable (including associated valves and piping) at any one time. If the system is not restored to meet the requirements of 3.4.2a within the time period specified, the reactor shall be placed in the hot shutdown condition. If the requirements of 3.4.2a are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition. Specification 3.0.1 applies to 3.4.2.b.

1. ONE emergency containment cooling unit may be out of service for a period of 24 hours. Prior to initiating maintenance the other TWO units shall be tested to demonstrate operability.

2. ONE containment spray pump may be out of service provided it is restored to operable status within 24 hours. Compliance with the requirements of Specification 4.0.3 shall be verified for the other pump.

3. Any valve in the system may be inoperable provided repairs are completed within 24 hours. Compliance with the requirements of Specification 4.0.3 shall be verified for all other valves that provide the duplicate function.

3. EMERGENCY CONTAINMENT FILTERING SYSTEM

a. The reactor shall not be made critical, except for low power physics tests unless:

1. THREE emergency containment filtering units are operable.

2. All valves, interlocks and piping associated with the above components and required for post-accident operation, are operable.

b. During power operation:

1. ONE unit may be inoperable for a period of 7 days if the other TWO are operable.

2. Any valve in the system may be inoperable provided repairs are completed within 7 days. Compliance with the requirements of Specification 4.0.3 shall be verified for all other valves that provide the duplicate function.

3. If after 7 days the unit is still inoperable, Specification 3.0.1 applies to 3.4.3.b.

4. COMPONENT COOLING SYSTEM

a. The reactor shall not be made critical, except for low power physics tests unless the following conditions are met:

1. THREE component cooling pumps are operable.

2. THREE component cooling heat exchangers are operable.

3. All valves, interlocks and piping associated with the above components are operable.

b. During power operation, the requirements of 3.4.4.a may be modified as stated below. If the sytem is not restored to meet the conditions of 3.4.4.a within the time period specified, the reactor shall be placed in the hot shutdown condition. If the requirements of 3.4.4.a are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition. Specification 3.0.1 applies to 3.4.4.b.

   1. ONE pump may be out of service for 7 days.

   2. a) A second pump may be out of service for a period of 24 hours. Compliance with the requirements of Specification 4.0.3 shall be verified for the third pump.

      b) ONE heat exchanger may be out of service for a period of 24 hours. Compliance with the requirements of Specification 4.0.3 shall be verified for the other heat exchangers.

5. <u>INTAKE COOLING WATER SYSTEM</u>

a. The reactor shall not be made critical unless the following conditions are met:

   1. THREE intake cooling water pumps and TWO headers are operable.

   2. All valves, interlocks and piping associated with the operation of these pumps, and required for post accident operation, are operable.

b. During power operation, the requirements of 3.4.5.a., above, may be modified to allow any one of the following components to be inoperable provided the remaining systems are in continuous operation. If the system is not restored to meet the requirements of 3.4.5.a. within the time period specified, the reactor shall be placed in the hot shutdown condition. If the requirements of 3.4.5.a are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition. Specification 3.0.1 applies to 3.4.5.b.

   1. One of the two headers may be out of service for a period of 24 hours. Compliance with the requirements of Specification 4.0.3 shall be verified for the remaining header.

   2. One intake cooling water pump may be out of service for a period of 24 hours. Compliance with the requirements of Specification 4.0.3 shall be verified for the other two pumps.

6. POST ACCIDENT CONTAINMENT VENT SYSTEM

a. The reactor shall not be made critical, except for low power physics tests unless:

1. The post accident containment vent system is operable.

2. All valves, interlocks, and piping associated with the above components and required for post-accident operation are operable.

b. During power operation:

1. The unit may be inoperable for a period of 7 days.

2. Any valve in the system may be inoperable provided repairs are completed within 7 days. Compliance with the requirements of Specification 4.0.3 shall be verified for all valves that provide the duplicate function.

3. If after 7 days the unit is still inoperable, Specification 3.0.1 applies to 3.4.6.b.

7. CONTROL ROOM VENTILATION

a. The reactor shall not be made critical, except for low power physics tests unless:

1. The control room ventilation system is operable.

2. All valves, interlocks, and piping associated with the above components and required for post-accident operation are operable.

b. During power operation:

1. The unit may be inoperable for a period of 3 1/2 days.

2. Any valve in the system may be inoperable provided repairs are completed within 3 1/2 days. Prior to initiating maintenance, all valves that provide the duplicate function shall be tested to demonstrate operability.

3. If after 3 1/2 days the unit is still inoperable, Specification 3.0.1 applies to 3.4.7.b.

3.8 STEAM AND POWER CONVERSION SYSTEMS

Applicability: Applies to the operating status of the steam and power conversion systems.

Objective: To define conditions of the steam-relieving capacity and auxiliary feedwater system.

Specification:

1. When the reactor coolant of a nuclear unit is heated above 350°F, the following conditions must be met:

   a. TWELVE (12) of its steam generator safety valves shall be operable (except for testing).

   b. System piping, interlocks and valves directly associated with the related components shall be operable.

   c. Its condensate storage tank shall contain a minimum of 185,000 gallons of water.

   d. Its main steam stop valves shall be operable in accordance with the requirements of Specification 4.0.3.

2. The iodine-131 activity on the secondary side of a steam generator shall not exceed 0.67 μCi/cc.

3. During power operation, if any of the conditions of 3.8.1 or 3.8.2 cannot be met within 48 hours, the reactor shall be shutdown and the reactor coolant temperature reduced below 350°F.

4. The following number of independent steam generator auxiliary feedwater pumps and associated flow path shall be operable in accordance with the requirements of Specification 4.10 when the reactor coolant is heated above 350°F:

   a. Single Nuclear Unit Operation

      Two auxiliary feedwater pumps capable of being powered from an operable steam supply.

   b. Dual Nuclear Unit Operation

      Three auxiliary feedwater pumps capable of being powered from an operable steam supply.

5. During power operation, if any of the conditions of 3.8.4 cannot be met, the reactor shall be shutdown and the reactor coolant temperature reduced below 350°F, unless one of the following conditions can be met:

a. For single unit operation with one of the two required auxiliary feedwater pumps inoperable, restore the inoperable pump to operable status within 72 hours or the reactor shall be shutdown and the reactor coolant temperature reduced below 350 F within the next 12 hours.

b. For dual unit operation with one of the three required auxiliary feedwater pumps inoperable, restore the inoperable pump to operable status within 72 hours or a reactor shall be shutdown and its reactor coolant temperature reduced below 350 F within the next 12 hours.

4.0 SURVEILLANCE REQUIREMENTS

4.0.1 Specified intervals may be adjusted plus or minus 25% to accommodate normal test schedules.

4.0.2 When the reactor is in a shutdown condition, some of the surveillance requirements discussed in this section are not required to be satisfied provided that the safety limits or limiting conditions for operation for the shutdown status are satisfied. When a surveillance activity is not completed because the reactor is shutdown and the surveillance is not required, the surveillance requirement shall be met prior to the time indicated in the applicable footnote.

4.0.3 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

a) Inservice inspection of ASME Code Class 1, 2, and 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 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

b) Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

c) The provisions of Specification 4.0.1 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d) Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e) Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

4.1 OPERATIONAL SAFETY REVIEW

<u>Applicability:</u> Applies to items directly related to safety limits and limiting conditions for operation.

<u>Objective:</u> To specify the minimum frequency and type of surveillance to be applied to equipment and conditions.

<u>Specification:</u> Calibration, testing, and checking of analog channels and testing of logic channels shall be performed as specified in Table 4.1-1.

Equipment and sampling tests shall be conducted as specified in Table 4.1-2.

TABLE 4.1-2 (Sheet 2 of 3)

MINIMUM FREQUENCIES FOR EQUIPMENT AND SAMPLING TESTS

	Check	Frequency	Max. Time Between Tests (Days)
5. Control Rods (cont'd)	Partial movement of full length rods	Biweekly while critical	20
6. Pressurizer Safety Valves	Tests performed in compliance with Specification 4.0.3		
7. Main Steam Safety Valves			
8. Containment Isolation Trip	Functioning	Each refueling shutdown	NA
9. Refueling System Interlocks	Functioning	Prior to each refueling	NA
10. Accumulator	Boron Concentration	At least once per 31 days and within 6 hours after each solution volume increase of $\geq$ 1% of tank volume.†	
11. Reactor Coolant System Leakage	Evaluate	Daily	NA
12. Diesel Fuel Supply	Fuel inventory	Weekly	10
13. Spent Fuel Pit	Boron Concentration	Prior to refueling	NA
14. Secondary Coolant	I-131 Concentration	Weekly*†	10
15. Vent Gas and Particulates	I-131 and Particulate Activity	Weekly*	10
16. Fire Protection Pump and Power Supply	Operable	Monthly	45
17. Turbine Stop and Control Valves, Reheater Stop and Intercept Valves	Closure	Monthly***	45
18. LP Turbine Rotor Inspector (w/o rotor disassembly)	V, MT, PT	Every 5 years	6 years
19. Spent Fuel Cask Crane Interlocks	Functioning	Within 7 days	7 days when crane is being used to maneuver spen fuel cask.

4.2 REACTOR COOLANT SYSTEM IN-SERVICE INSPECTION

Applicability: Applies to pre-operational and in-service structural surveillance of the reactor coolant system boundary.

Objective: To assure the continued integrity of the reactor coolant system boundary.

Specification:

4.2.1 a. Except as listed below, there are no additional surveillance requirements other than those required by Specification 4.0.3.

4.2.2 The inspection interval shall be 10 years.

4.2.3 In addition to the requirements of Specification 4.0.3 each reactor coolant pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory guide 1.14, Revision 1, August 1975.

4.2.4 Irradiation specimens shall be examined by Tensile and Charpy V Notch methods. Capsule 1 shall be removed and examined at the first region replacement. Capsule 2 shall be removed and examined at the fourth region replacement. Capsule 3 shall be removed and examined after twenty years of operation. Capsule 4 shall be removed and examined after thirty years of operation. Capsule 5 shall be removed and examined after forty years of operation.

4.2.5 STEAM GENERATOR INSPECTION

4.2.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.2-2.

4.2.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.2-3. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.2.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.2.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas.

b. The sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

   1. All nonplugged tubes that previously had detectable wall penetrations (greater than 20%), and

   2. Tubes in those areas where experience has indicated potential problems.

   3. A tube inspection (pursuant to Specification 4.2.5.4.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples in the inservice inspection may be less than a full tube inspection by concentrating (selecting at least 50% of the tubes to be inspected) the inspection on those areas of the tube sheet array and on those portions of tubes where tubes with imperfections were previously found.

The results of each sample inspection shall be classified into one of the following three categories:

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

NOTE: In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

4.2.5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

a. The first inservice inspection shall be performed after six effective full power months of operation but within 24 calendar months following replacement of steam generators. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the inservice inspection of a steam generator conducted in accordance with Table 4.2-3 requires a third sample inspection whose results fall in Category C-3, the inspection frequency shall be reduced to at least once per 20 months. The reduction in inspection frequency shall apply until a subsequent inspection demonstrates that a third sample inspection is not required.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.2-3 during the shutdown subsequent to any of the following conditions:

1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.1.3.g.

2. A seismic occurrence greater than the operating Basis Earthquake (OBE).

3. A loss-of-coolant accident resulting in rapid depressurization of the primary system, or

4. A main steam line or feedwater line break resulting in rapid depressurization of the affected steam generator.

4.2.5.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing, indications below 20% of the nominal tube wall thickness, if detectable may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection or such severity that it exceeds the plugging limit. A tube containing a defect is defective. Any tube which does not permit the passage of the eddy-current inspection probe shall be deemed a defective tube.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service because it may become unserviceable prior to the next inspection and is equal to 40% of the nominal tube wall thickness.

7. Unserviceable described the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of OBE, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.2.5.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.2-3.

4.2.5.5 Reports

a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days.

b. The complete results of the steam generator tube inservice inspection shall be included in the Annual Changes, Tests and Experiment Reports for the period in which this inspection was completed. This report shall include:

   1. Number and extent of tubes inspected.

   2. Location and percent of wall-thickness penetration for each indication of an imperfection.

   3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 and require prompt notification of the Commission shall be reported pursuant to Specification 6.9.2.a prior to resumption of plant operation. The written followup of this report shall provide a description of investigations conducted to determine the cause of the tube degradation and corrective measures taken to prevent recurrence.

TABLE 4.2-1 WAS DELETED IN ITS ENTIRETY

4.4.2 LOCAL PENETRATION TESTS

Test Procedure and Frequency

Local leak detection tests of the following components shall be performed at a pressure not less than 50 psig using pressure decay, soap bubble, halogen detection or equivalent methods at the frequency listed, unless otherwise noted:

1. Containment purge valves (pressure applied in connecting duct) - each refueling.

2. Personnel and Emergency Airlocks

   a. *Within 3 days of every first of a series of openings when containment integrity is required, verify that door seals have not been damaged or seated improperly by vacuum testing the volume between the door seals in accordance with approved plant procedures.

   b. At least once per 6 months, conduct an overall airlock leakage test to verify that the overall airlock leakage rate is within its limit.

3. Equipment access opening (pressure applied between gaskets) - annually and after use.

4. Fuel transfer tube flange (pressure applied between gaskets) - each refuleing.

5. Electrical penetrations (pressure applied to canister) - each refueling.

Acceptance Criteria:

Repairs and tests shall be made whenever the sum of the local leak rate tests, including the isolation valves discussed in 4.4.3, exceeds sixty percent of the total containment allowable leak rate.

4.4.3 ISOLATION VALVES

Containment isolation valves shall be tested in accordance with 10 CFR 50, Appendix J.

No additional surveillance requirements other than those required by Specification 4.0.3.

4.4.4 RESIDUAL HEAT REMOVAL SYSTEM

No additional surveillance requirements other than those required by Specification 4.0.3.

4.5 SAFETY INJECTION

Applicability: Applies to testing of the Safety Injection System.

Objective: To verify that the subject systems will respond promptly and perform their design functions.

Specification:

1. SYSTEM TESTS

   a. System tests shall be performed at each refueling shutdown. The test shall be performed in accordance with the following procedure:

      With the Reactor Coolant System pressure equal to or less than 350 psig and temperature equal to or less than 350 F, a test safety injection signal will be applied to initiate operation of the system. The breakers for the residual heat removal pump motors will be tested either in the test position or by actual residual heat removal pump motor operation resulting from the test safety injection signal.

   b. The test will be considered satisfactory if control panel indication and visual observations indicate that all components have received the safety injection signal in the proper sequence and timing, appropriate breakers shall open and close, and all automatic valves shall complete their travel.

2. COMPONENT TESTS

   a. Pumps

      No additional surveillance requirements other than those required by Specification 4.0.3.

   b. Valves

      No additional surveillance requirements other than those required by Specification 4.0.3.

2. COMPONENT TESTS

a. Pumps

For the containment spray pumps, there are no additional surveillance requirements other than those required by Specification 4.0.3.

b. Fans

Emergency Containment Cooling fans shall be started at intervals not greater than one (1) month.††

Acceptable levels of performance shall be that the fan motors reach their nominal operating current for the containment atmosphere during the test, and operate for at least fifteen minutes.

c. Valves

No additional surveillance requirements other than those required by Specification 4.0.3.

†† - N.A. during cold or refueling shutdowns. The specified tests, however, shall be performed within one surveillance interval prior to heatup above 200 F.

2. POST ACCIDENT CONTAINMENT VENT SYSTEM

1. Operating Tests

a. Operating tests shall be performed during refueling but not longer than 18 months. The tests shall consist of visual inspection of the system and pressure drop and air flow measurements. Visual inspection shall include a search for any foreign materials and gasket deterioration in the HEPA filters and charcoal adsorbers. Less than 6" of water pressure drop at 55 cfm flow shall constitute acceptable performance.

b. Valves

No additional surveillance requirements other than those required by Specification 4.0.3.

2. Performance Tests

a. A visual inspection of the sytem shall be made before each DOP test and halogenated hydrocarbon leak test. At least once per 18 months or after 720 hours of system operation, in-place DOP and halogenated hydrocarbon tests at design flow (55 cfm ± 10%) and carbon analysis, or carbon replacement, for the Post Accident Containment Vent filters shall be performed. In addition, carbon analysis (or carbon replacement), DOP and halogenated hydrocarbon tests at design flow (55 cfm ± 10%) shall be performed after (1) any structural maintenance on system housings which might have affected filter bank efficiency, (2) after complete or partial replacement of a filter bank or (3) after exposure of the filters to effluents from painting, fire or chemical release. Removal of ≥99% DOP and ≥99% halogenated hydrocarbon shall constitute acceptable performance.

b. Laboratory carbon sample analysis shall show ≥90% methyl radio-iodine removal or the charcoal shall be replaced with charcoal that meets or exceeds the criteria of position C.6.a of Regulatory Guide 1.52 (Revision 2). The sample shall be taken in accordance with position C.6.b of Regulatory Guide 1.52. Carbon analysis shall be performed in accordance with ANSI N510-1975. Analysis shall verify the above removal efficiency for radio-iodine within 45 days after removal of the sample.

c. The hydrogen concentration measurement instrument shall be calibrated with proper consideration for humidity.

3. CONTROL ROOM VENTILATION (EMERGENCY INTERNAL CLEANUP) SYSTEM

1. A visual inspection shall be made before each in-place DOP test, halogenated hydrocarbon leak test, and airflow distribution test. The Control Room Ventilation System shall be operated monthly for at least 15 minutes to demonstrate operability. Auto initiation of the systems operations shall be checked during refueling, but not longer than 18 months. Pressure drop measurements across the filter bank shall be made annually. Less than 6" of water pressure drop at designed flow (1,000 cfm ± 10%) across the combined HEPA filter and charcoal adsorbers shall constitute acceptable performance. A visual inspection shall include a search for any foreign materials and gasket deterioration in the HEPA filters and charcoal adsorbers.

2. Performance Tests

   a. A visual inspection shall be made before each in-place DOP test, halogenated hydrocarbon leak test and airflow distribution test. At least once per 18 months or after 720 hours of system operation, in-place DOP and halogenated hydrocarbon tests at design flow (1,000 cfm ± 10%) and carbon analysis shall be performed after (1) any structural maintenance on system housings, which might have affected filter bank efficiency, (2) after complete or partial replacement of a filter bank, or (3) after operational exposure

4.9 MAIN STEAM ISOLATION VALVES

<u>Applicability:</u> Applies to periodic testing of the main steam isolation valves.

<u>Objective:</u> To verify the ability of the main steam isolation valves to close upon signal.

<u>Specification:</u> Each main steam isolation valve shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.3.

4.10 AUXILIARY FEEDWATER SYSTEM

Applicability: Applies to periodic testing requirements of the auxiliary feedwater system.

Objective: To verify the operability of the auxiliary feedwater system and its ability to respond properly when required.#*

Specification:

1. Each turbine-driven auxiliary feedwater pump shall be tested in accordance with the requirements of Specification 4.0.3 and a flow rate of 373 gpm established to the steam generators.

2. Part of the tests acceptance criteria shall be considered satisfactory if control panel indication and visual observation of the equipment demonstrate that all components have operated properly.

3. At least once per 18 months:

   (a) Verify that each automatic valve in the flow path actuates to its correct position upon receipt of each auxiliary feedwater actuation test signal.

   (b) Verify that each auxiliary feedwater pump receives a start signal as designed automatically upon receipt of each auxiliary feedwater actuation test signal.

* N.A. during cold or refueling shutdowns (only for the unit at cold or refueling shutdown). The specified tests, however, shall be performed within one surveillance interval prior to starting the turbine.

\# NOTE: If any local manual realignment of valves is required when operating the Auxiliary Feedwater pumps, a dedicated individual, who is in communication with the control room, shall be stationed at the auxiliary feedwater pump area. Upon instructions from the control room, this operator would realign the valves in the AFW system train to its normal operational alignment.

B4.0 BASES FOR SURVEILLANCE REQUIREMENTS

4.0.1 This specification provides a grace period so that intervals may be adjusted to accommodate normal test schedules.

4.0.2 This specification provides guidance for surveillance requirements when the reactor is in a shutdown condition.

4.0.3 This specification ensures that inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these Technical Specifications.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Code and applicable Addenda.

.B4.2 BASES FOR REACTOR COOLANT SYSTEM IN-SERVICE INSPECTION

This specification ensures that inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. This is accomplished by referencing the inspections required by Specification 4.0.3.

MISCELLANEOUS INSPECTIONS

Reactor Coolant Pump Flywheels

The flywheels shall be visually examined per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14 Revision 1, August 1975.

Materials Irradiation Surveillance Specimens

The reactor vessel surveillance program includes eight specimen capsules to evaluate radiation damage based on pre-irradiation and post-irradiation testing of specimens. The specimens are located about 3" from the vessel wall at the axial midplane.

Capsule No. 1 is scheduled to be removed at the first region replacement. The exposure of this capsule leads the vessel maximum exposure by a factor of 2.1. Thus, this capsule provides information for approximately a four-year exposure to the vessel.

Capsule No. 2 is scheduled to be removed at the fourth region replacement. This capsule leads the vessel maximum exposure by a factor of 0.8 and thus will provide data for a four-year exposure to the vessel. This sample also contains weld metal which is not present in Capsule No. 1.

Capsule No. 3 leads the vessel maximum exposure by a factor of 2.2 and is scheduled to be removed after twenty years. Thus, sample No. 3 will provide data for an exposure to the vessel of approximately forty years.

Capsules No. 4 and 5 lead the maximum vessel exposure by factors of 0.7 and 0.5, respectively. Thus, Capsule No. 4, which is scheduled to be removed after thirty years, provides data for a vessel exposure of twenty-one years and Capsule No. 5, which is scheduled to be removed at forty years, provides data for a vessel exposure of twenty years.

In addition to the capsules discussed above, there are three spares.

Steam Generator Tube Inspection

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. In service inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those parameter limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these parameter limits, localized corrosion may likely result in stress corrosion racking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 1 gallon per minute, total). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 1 gallon per minute can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with the all volatile treatment (AVT) of secondary coolant. However, even if a defect of similar type should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required of all tubes with imperfections exceeding the plugging limit which, by the definition of Specification 4.2.5.4.a is 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing in-service inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.2.a prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.