## 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 basline 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 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC.

4.2.3 The structural integrity of the reactor coolant system boundary shall be maintained at the level required by the original acceptance standards throughout the life of the station. Any evidence, as a result of the tests outlined in the Table IS-261 of Section XI of the code, that defects have developed or grown, shall be investigated.

4.2.4 To assure the structural integrity of the reactor internals throughout the life of the unit, the two sets of main internals bolts (connecting the core barrel to the core support, shield and to the lower grid cylinder) shall remain in place and under tension. This will be verified by visual inspection to determine that the welded bolt locking caps remain in place. All locking caps will be inspected after hot functional testing and whenever the internals are removed from the vessel during a refueling or maintenance shutdown. The core barrel to core support shield caps will be inspected each refueling shutdown.

4.2.5 Sufficient records of each inspection shall be kept to allow comparison and evaluation of future inspections.

4.2.6 Surface and volumetric examination of the reactor coolant pump flywheels will be conducted coincident with refueling or maintenance shutdowns such that during 10 year intervals all four reactor coolant pump flywheels will be examined. Such examinations will be performed to the extent possible through the access ports, i.e., those areas of the flywheel accessible without motor disassembly. The surface and volumetric examination may be accomplished by Acoustic Emission Examination as an initial examination method. Should the results of the Acoustic Emission Examination indicate that additional examination is necessary to ensure the structural integrity of the flywheel, then other appropriate NDE methods will be performed on the area of concern.

Bases

The surveillance program has been developed to comply with the applicable edition of Section XI and addenda of the ASME Boiler and Pressure Vessel Code, Inservice Inspection of Nuclear Reactor Coolant Systems, as required by 10CFR50.55a, to the extent practicable within limitations of design, geometry and materials of construction.

## 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 2155 psig, prior to the reactor being made critical, in accordance with the ASME Boiler and Pressure Vessel Code, Section XI; IWA-5000.

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. The ASME Boiler and Pressure Vessel Code, Section XI; IWA-5000 requires a system leak test at nominal operating pressure (2155 psig) following system opening. At the end of refueling outages, this test also satisfies the requirements of IWB-2500, Table IWB-2500-1; Category B-P items B15-10, B15-20, B15-30, B15-40, B15-50, B15-60, and B15-70 for all Class I pressure retaining components.

REFERENCES

(1) FSAR, Section 4
(2) ASME Boiler and Pressure Vessel Code, Section XI

## 4.4 REACTOR BUILDING

### 4.4.1 Reactor Building Leakage Tests

Applicability

Applies to the reactor building.

Objective

To verify that leakage from the reactor building is maintained within allowable limits.

Specification

4.4.1.1 Integrated leakage rate tests shall be conducted and visual inspections performed in accordance with the Reactor Building Leakage Rate Testing Program.

4.4.1.1.1 Deleted

4.4.1.1.2 Deleted

4.4.1.1.3 Deleted

4.4.1.1.4 Integrated leakage rate testing frequencies shall be in accordance with the Reactor Building Leakage Rate Testing Program.

4.4.1.1.5 Deleted

4.4.1.1.6 Deleted

4.4.1.1.7 Deleted

4.4.1.2 Local leakage rate tests shall be conducted in accordance with the Reactor Building Leakage Rate Testing Program.

4.4.1.2.1 Deleted

4.4.1.2.2 Deleted

4.4.1.2.3 Deleted

4.4.1.2.4 Deleted

4.4.1.2.5 Local leakage rate testing frequencies shall be in accordance with the Reactor Building Leakage Rate Testing Program.

4.4.1.3 Deleted

4.4.1.4 Isolation Valve Functional Tests

Every three months, 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 plant operation. The latter valves shall be tested once every 18 months.

4.4.1.5 Deleted

Bases (1)

The reactor building is designed for an internal pressure of 59 psig and a steam-air mixture temperature of 285°F.

The peak calculated reactor building pressure for the design basis loss of coolant accident, $P_a$, is 54 psig. The maximum allowable reactor building leakage rate, $L_a$, shall be 0.20% of containment air weight per day at $P_a$.

The reactor building will be periodically leakage tested in accordance with the Reactor Building Leakage Rate Testing Program. These periodic testing requirements verify the reactor building leakage rate does not exceed the assumptions used in the safety analysis. At $\leq 1.0\ L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and Type C leakage, and $\leq 0.75\ L_a$ for overall Type A leakage. At all other times between required leakage tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$.

REFERENCE

(1) FSAR, Sections 5 and 13.

## 4.5 EMERGENCY CORE COOLING SYSTEM AND REACTOR BUILDING COOLING SYSTEM PERIODIC TESTING

### 4.5.1 Emergency Core Cooling Systems

Applicability

Applies to periodic testing requirement for emergency core cooling systems.

Objective

To verify that the emergency core cooling systems are operable.

Specification

4.5.1.1 System Tests

4.5.1.1.1 High Pressure Injection System

(a) Once every 18 months, a system test shall be conducted to demonstrate that the system is operable. A test signal will be applied to demonstrate actuation of the high pressure injection system for emergency core cooling operation.

(b) The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly; all appropriate pump breakers shall have opened or closed and all valves shall have completed their travel.

4.5.1.1.2 Low Pressure Injection System

(a) Once every 18 months, a system test shall be conducted to demonstrate that the system is operable. The test shall be performed in accordance with the procedure summarized below:

(1) A test signal will be applied to demonstrate actuation of the low pressure injection system for emergency core cooling operation.

(2) Verification of the engineered safeguard function of the service water system which supplies cooling water to the decay heat removal coolers shall be made to demonstrate operability of the coolers.

(b) The test will be considered satisfactory if control board indication verifies that all components have responded to the actuation signal properly; all appropriate pump breakers shall have opened or closed, and all valves shall have completed their travel.

4.5.1.1.3 Core Flooding System

(a) Once every 18 months, a system test shall be conducted to demonstrate proper operation of the system. During this test, verification shall be made that the check valves in the core flooding tank discharge lines operate properly.

(b) The test will be considered satisfactory if control board indication of core flood tank level verifies that all check valves have opened.

4.5.1.2 Component Tests

4.5.1.2.1 Pumps

Approximately quarterly, the high pressure and low pressure injection pumps shall be started and operated to verify proper operation. Acceptable performance will be indicated if the pump starts, operates for fifteen minutes, and the discharge pressure and flow are within ±10% of the initial level of performance as determined using test flow paths.

4.5.1.2.2 Valves - Power Operated

(a) At intervals not to exceed three months, each engineered safety feature valve in the emergency core cooling systems and each engineered safety feature valve associated with emergency core cooling in the service water system which are designed to open in the event of a LOCA shall be tested to verify operability.

(b) The acceptable performance of each power operated valve will be that motion is indicated upon actuation by appropriate signals.

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.

With the reactor shutdown, the check valves in each core flooding line are checked for operability by reducing the reactor coolant system pressure until the indicated level in the core flood tanks verify that the check valves have opened.

<u>REFERENCE</u>

FSAR Section 6

### 4.5.2 Reactor Building Cooling Systems

**Applicability**

Applies to testing of the reactor building emergency cooling systems.

**Objective**

To verify that the reactor building emergency cooling systems are operable.

**Specification**

4.5.2.1 System Tests

4.5.2.1.1 Reactor Building Spray System

(a) Once every 18 months, a system test shall be conducted to demonstrate proper operation of the system. A test signal will be applied to demonstrate actuation of the reactor building spray system (except for reactor building inlet valves to prevent water entering nozzles).

(b) Station compressed air or smoke will be introduced into the spray headers to verify the availability of the headers and spray nozzles at least every five years.

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

4.5.2.1.2 Reactor Building Cooling System

(a) At least once per 14 days, each reactor building emergency cooling train shall be tested to demonstrate proper operation of the system. The test shall be performed in accordance with the procedure summarized below:

(1) Verifying a service water flow rate of ≥ 1200 gpm to each train of the reactor building emergency cooling.[1]

(2) Addition of a biocide to the service water during the surveillance in 4.5.2.1.2.a.1 above, whenever service water temperature is between 60F and 80F.

(b) At least once per 31 days, each reactor building emergency cooling train shall be tested to demonstrate proper operation of the system. The test shall be performed in accordance with the procedure summarized below:

(1) Starting (unless already operating) each operational cooling fan from the control room.

---

[1] Surveillance Requirement 4.5.2.1.2(a)(1) will not be performed on the green train of the reactor building emergency cooling system until cooling fan VSF-1D is repaired and the green train is returned to normal configuration. This note will remain in effect until July 14, 1995.

(2) Verifying that each operational cooling fan operates for at least 15 minutes.

(c) Once every 18 months, a system test shall be conducted to demonstrate proper operation of the system. The test shall be performed in accordance with the procedure summarized below:

(1) A test signal will be applied to actuate the reactor building emergency cooling operation.

(2) Verification of the engineered safety features function of the service water system which supplies the reactor building emergency coolers shall be made to demonstrate operability of the coolers.

(3) 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

At intervals not to exceed 3 months the reactor building spray pumps shall be started and operated to verify proper operation. Acceptable performance will be indicated if the pump starts, operates for fifteen minutes, and the discharge pressure and flow are within ±10% of a point on the pump head curve.

4.5.2.2.2 Valves

At intervals not to exceed three months each engineered safety features valve in the reactor building spray and reactor building emergency cooling system and each engineered safety features valve associated with reactor building emergency cooling in the service water system shall be tested to verify that it is operable.

Bases

The reactor building emergency cooling system and reactor building spray system are redundant to each other in providing post-accident cooling of the reactor building atmosphere to prevent the building pressure from exceeding the design pressure. As a result of this redundancy in cooling capability, the allowable out of service time requirements for the reactor building emergency cooling system have been appropriately adjusted. However, the allowable out of service time requirements for the reactor building spray system have been maintained consistent with that assigned other inoperable engineered safeguard equipment since the reactor building spray system also provides a mechanism for removing iodine from the reactor building atmosphere.



effective May 10, 1991

Addition of a biocide to service water is performed during reactor building emergency cooler surveillance to prevent buildup of Asian clams in the coolers when service water is pumped through the cooling coils. This is performed when service water temperature is between 60F and 80F since in this water temperature range Asian clams can spawn and produce larva which could pass through service water system strainers.

The delivery capability of one reactor building spray pump at a time can be tested by opening the valve in the line from the borated water storage tank, opening the corresponding valve in the test line, and starting the corresponding pump. Pump discharge pressure and flow indication demonstrate performance.

With the pumps shut down and the borated water storage tank outlet closed, the reactor building spray injection valves can each be opened and closed by operator action. With the reactor building spray inlet valves closed, low pressure air or 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 emergency cooling system are arranged so that they can be visually inspected. The cooling fans and coils 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.

Two service water pumps are normally operating. At least once per month operation of one pump is shifted to the third pump, so testing will be unnecessary.

As the reactor building fans are normally operating, starting for testing is unnecessary for those verified to be operating.

<u>Reference</u>

FSAR, Section 6



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## 4.6 AUXILIARY ELECTRICAL SYSTEM TESTS

### Applicability

Applies to the periodic testing and surveillance requirements of the auxiliary electrical system to ensure it will respond promptly and properly when required.

### Specification

4.6.1 Diesel Generators

1. Each diesel generator shall be manually started each month and demonstrated to be ready for loading within 15 seconds. The signal initiating the start of the diesel shall be varied from one test to another (start with handswitch at control room panel and at diesel local control panel) to verify all starting circuits are operable. The generator shall be synchronized from the control room and loaded to full rated load and allowed to run until diesel generator operating temperatures have stabilized.

2. A test shall be conducted once every 18 months to demonstrate the ability of the diesel generators to perform as designed by:

   a. simulating a loss of off-site power,

   b. simulating of loss of off-site power in conjunction with an ESF signal,

   c. simulating interruption of off-site power and subsequent reconnection of the on-site power source to their respective busses, and

   d. operating the diesel generator for ≥1 hour after operating temperatures have stabilized.

3. Each diesel generator shall be given an inspection once every 18 months following the manufacturer's recommendations for this class of standby service. (A one-time extension of this interval is allowed so that these may be performed during the 1R9 refueling outage, and completed no later than December 1, 1990.)

4. During the monthly diesel generator test specified in paragraph 1 above, the following shall be performed:

   a. The diesel generator starting air compressors shall be checked for operation and their ability to recharge the air receivers.

   b. The diesel oil transfer pumps shall be checked for operability and their ability to transfer oil to the day tank.

   c. The day tank fuel level shall be verified.

   d. The emergency storage tank fuel level shall be verified.

e. Diesel fuel from the emergency storage tank shall be sampled and found to be within acceptable limits specified in Table 1 of ASTM D975-68 when checked for viscosity, water, and sediment.

5. Once every 31 days the pressure in the required starting air receiver tanks shall be verified to be ≥ 175 psig.

Once every 18 months, the capacity of each diesel oil transfer pump shall be verified to be at least 10 gpm.

4.6.2 DC Sources and Battery Cell Parameters

1. Verify battery terminal voltage is ≥ 124.7 V on float charge once each 7 days.

2. Verify battery capacity is adequate to supply, and maintain in operable status, the required emergency loads for the design duty cycle when subjected to either a battery service test or a modified performance discharge test once every 18 months.

3. Verify battery capacity is ≥ 80% of the manufacturers rating when subjected to a performance discharge test or a modified performance discharge test once every 60 months, once every 24 months when battery has reached 85% of the service life with capacity ≥ 100% of the manufacturers rating and showing no degradation, and once every 12 months when battery shows degradation or has reached 85% of the service life and capacity is < 100% of the manufacturer's rating.

4. Any battery charger which has not been loaded while connected to its 125V d-c distribution system for at least 30 minutes during every quarter shall be tested and loaded while connected to its bus for 30 minutes.

5. Verify battery pilot cell parameters meet Table 4.6-1 Category A limits once per 7 days.

6. Verify average electrolyte temperature of representative cells is ≥ 60°F once per 92 days.

7. Verify battery cell parameters meet Table 4.6-1 Category B limits once per 92 days and once within 24 hours after a battery discharge to < 110 V and once within 24 hours after a battery overcharge to > 145 V.

8. Verify electrolyte temperature of pilot cell is ≥ 60°F once per 31 days.

4.6.3 Emergency Lighting

The correct functioning of the emergency lighting system shall be verified once every 18 months.

Table 4.6-1 (page 1 of 1)
Battery Cell Surveillance Requirements

| PARAMETER | CATEGORY A: LIMITS FOR EACH DESIGNATED PILOT CELL | CATEGORY B: LIMITS FOR EACH CONNECTED CELL | CATEGORY C: ALLOWABLE LIMITS FOR EACH CONNECTED CELL |
|---|---|---|---|
| Electrolyte Level | > Minimum level indication mark, and ≤ 1/4 inch above maximum level indication mark[a] | > Minimum level indication mark, and ≤ 1/4 inch above maximum level indication mark[a] | Above top of plates, and not overflowing |
| Float Voltage | ≥ 2.13 V | ≥ 2.13 V | > 2.07 V |
| Specific Gravity[b] [c] | ≥ 1.195 | ≥ 1.190<br>AND<br>Average of all connected cells > 1.195 | Not more than 0.020 below average connected cells<br>AND<br>Average of all connected cells ≥ 1.190 |

(a) It is acceptable for the electrolyte level to temporarily increase above the specified maximum during equalizing charges provided it is not overflowing.

(b) Corrected for electrolyte temperature.

(c) A battery charging current of < 2 amps when on float charge is acceptable for meeting specific gravity limits following a battery recharge, for a maximum of 7 days. When charging current is used to satisfy specific gravity requirements, specific gravity of each connected cell shall be measured prior to expiration of the 7 day allowance.

**Bases**

The emergency power system provides power requirements for the engineered safety features in the event of a DBA. Each of the two diesel generators is capable of supplying minimum required engineered safety features from independent buses. This redundancy is a factor in establishing testing intervals. The monthly tests specified above will demonstrate operability and load capacity of the diesel generator. The fuel supply and diesel starter motor air pressure are continuously monitored and alarmed for abnormal conditions. Starting on complete loss of off-site power will be verified by simulated loss-of-power tests once every 18 months.

The SR 4.6.2.1 verification of battery terminal voltage while on float charge helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. Float charge is the condition in which the battery charger is supplying the continuous charge required to overcome the internal losses of a battery and maintain the battery in a fully charged state. The voltage requirements are based on the nominal design voltage of the battery (2.15 V per cell average) and are consistent with the battery vendor allowable minimum volts per cell limits. The inability to meet this requirement constitutes an inoperable battery.

The SR 4.6.2.2 battery service test is a special test of the battery capability, as found, to satisfy the design requirements (battery duty cycle) of the DC electrical power system. The discharge rate and test length should correspond to the design duty cycle requirements. A modified performance discharge test may be performed in lieu of a service test. The inability to meet this requirement constitutes an inoperable battery.

The modified performance discharge test is a simulated duty cycle consisting of just two rates; the one minute rate published for the battery or the largest current load of the duty cycle, followed by the test rate employed for the performance test, both of which envelope the duty cycle of the battery. Since the ampere-hours removed by a rated one minute discharge represents a very small portion of the battery capacity, the test rate can be changed to that for the performance test without compromising the results of the performance discharge test. The battery terminal voltage for the modified performance discharge test should remain above the minimum battery voltage specified in the battery service test for the duration of time equal to that of the service test.

A modified performance discharge test is a test of the battery capacity and its ability to provide a high rate, short duration load (usually the highest rate of the duty cycle). This will often confirm the battery's ability to meet the critical period of the load duty cycle, in addition to determining its percentage of rated capacity. Initial conditions for the modified performance discharge test should be identical to those specified for a service test and the test discharge rate must envelope the duty cycle of the service test if the modified performance discharge test is performed in lieu of a service test.

The SR 4.6.2.3 battery performance discharge test is a test of constant current capacity of a battery after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage. The inability to meet this requirement constitutes an inoperable battery.

Either the battery performance discharge test or the modified performance discharge test, described above, is acceptable for satisfying SR 4.6.2.3; however, only the modified performance discharge test may be used to satisfy SR 4.6.2.3 while satisfying the requirements of SR 4.6.2.2 at the same time.

The acceptance criteria for this surveillance are consistent with IEEE-450. This reference recommends that the battery be replaced if its capacity is below 80% of the manufacturer's rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements.

The frequency for this test is normally 60 months. If the battery shows signs of degradation, or if the battery has reached 85% of its service life and capacity is < 100% of the manufacturer's rating, the frequency is reduced to 12 months. However, if the battery shows no degradation but has reached 85% of its service life, the frequency is only reduced to 24 months for batteries that retain ≥ 100% of the manufacturer's ratings. Degradation is indicated, according to IEEE-450, when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is ≥ 10% below the manufacturer's rating.

SR 4.6.2.4 requires that each required battery charger be capable of supplying the connected loads while maintaining the battery fully charged. This is based on the assumption that the batteries are fully charged at the beginning of a design basis accident, and on the safety function of providing adequate power for the design basis accident loads.

SR 4.6.2.5 verifies that the Table 4.6-1 Category A battery cell parameters are consistent with vendor recommendations and IEEE-450, which recommend regular battery inspections (at least once per month) including voltage, specific gravity, and electrolyte level of pilot cells.

The SR 4.6.2.6 verification that the average temperature of representative cells is ≥ 60°F is consistent with a recommendation of IEEE-450, which states that the temperature of electrolytes in representative cells (~10% of all connected cells) should be determined on a quarterly basis. Lower than normal temperatures act to inhibit or reduce battery capacity. This surveillance ensures that the operating temperatures remain within an acceptable operating range. This limit is based on manufacturer recommendations.

SR 4.6.2.7 verifies that the Table 4.6-1 Category B battery cell parameters are consistent with vendor recommendations and IEEE-450, which recommend regular battery inspections (at least once per quarter) including voltage, specific gravity, and electrolyte level of each connected cell. In addition, within 24 hours after a battery discharge to < 110 V or a battery overcharge to > 145 V, the battery must be demonstrated to meet Category B limits. Transients, such as motor starting transients, which may momentarily cause battery voltage to drop to ≤ 110 V, do not constitute a battery discharge provided battery terminal voltage and float current return to pre-transient values. This inspection is also consistent with IEEE-450, which recommends special inspections following a severe discharge or overcharge, to ensure that no significant degradation of the battery occurs as a consequence of such discharge or overcharge.

The SR 4.6.2.8 verification that the temperature of the pilot cell is ≥ 60°F is consistent with a recommendation of IEEE-450, which states that the temperature of electrolytes in pilot cells should be determined on a monthly basis. Lower than normal temperatures act to inhibit or reduce battery capacity. This surveillance ensures that the operating temperatures remain within an acceptable operating range. This limit is based on manufacturer recommendations.

Table 4.6-1 delineates the limits on electrolyte level, cell float voltage, and specific gravity for three different categories. The meaning of each category is discussed below.

Category A defines the normal parameter limit for each designated pilot cell in each battery. The cells selected as pilot cells are those whose temperature, voltage and electrolyte specific gravity approximate the state of charge of the entire battery.

The Category A limits specified for electrolyte level are based on manufacturer recommendations and are consistent with the guidance in IEEE-450, with the extra 1/4 inch allowance above the high water level indication for operating margin to account for temperatures and charge effects. In addition to this allowance, footnote (a) to Table 4.6-1 permits the electrolyte level to be above the specified maximum level during equalizing charge, provided it is not overflowing. These limits ensure that the plates suffer no physical damage and that adequate electron transfer capability is maintained in the event of transient conditions. IEEE-450 recommends that electrolyte level readings should be made only after the battery has been at float charge for at least 72 hours.

The Category A limit specified for float voltage is ≥ 2.13 V per cell. This value is based on the battery vendor allowable minimum cell voltage and on a recommendation of IEEE-450, which states that prolonged operation of cells < 2.13 V can reduce the life expectancy of cells.

The Category A limit specified for specific gravity for each pilot cell is ≥ 1.195. This value is characteristic of a charged cell with adequate capacity. According to IEEE-450, the specific gravity readings are based on a temperature of 77°F (25°C).

The specific gravity readings are corrected for actual electrolyte temperature. For each 3°F (1.67°C) above 77°F (25°C), 1 point (0.001) is added to the reading; 1 point is subtracted for each 3°F below 77°F. The specific gravity of the electrolyte in a cell increases with a loss of water due to electrolysis or evaporation.

Category B defines the normal parameter limits for each connected cell. The term "connected cell" excludes any battery cell that is jumpered out.

The Category B limits specified for electrolyte level and float voltage are the same as those specified for Category A and have been discussed above. The Category B limit specified for specific gravity for each connected cell is ≥ 1.190 with the average of all connected cells > 1.195. These values are based on manufacturer's recommendations. The minimum specific gravity value required for each cell ensures that the effects of a highly charged or newly installed cell will not mask overall degradation of the battery.

Category C defines the limits for each connected cell. These values, although reduced, provide assurance that sufficient capacity exists to perform the intended function and maintain a margin of safety. When any battery parameter is outside the Category C limits, the assurance of sufficient capacity described above no longer exists and the battery must be declared inoperable.

The Category C limits specified for electrolyte level (above the top of the plates and not overflowing) ensure that the plates suffer no physical damage and maintain adequate electron transfer capability. The Category C limit for float voltage is consistent with IEEE-450, which states that a cell voltage of 2.07 V or below, under float conditions and not caused by elevated temperature of the cell, indicates internal cell problems and may require cell replacement.

The Category C limit of average specific gravity ≥ 1.190 is based on manufacturer recommendations. In addition to that limit, it is required that the specific gravity for each connected cell must be no less than 0.020 below the average of all connected cells. This limit ensures that the effect of a highly charged or new cell does not mask overall degradation of the battery.

Footnotes (b) and (c) to Table 4.6-1 are applicable to Category A, B, and C specific gravity. Footnote (b) to Table 4.6-1 requires the above mentioned correction for electrolyte temperature. The value of 2 amps used in footnote (c) is the nominal value for float current established by the battery vendor as representing a fully charged battery with an allowance for overall battery condition. This current provides, in general, an indication of overall battery condition.

Because of specific gravity gradients that are produced during the recharging process, delays of several days may occur while waiting for the specific gravity to stabilize. A stabilized charger current is an acceptable alternative to specific gravity measurement for determining the state of charge. This phenomenon is discussed in IEEE-450. Footnote (c) to Table 4.6-1 allows the float charge current to be used as an alternate to specific gravity for up to 7 days following a battery recharge. Within 7 days each connected cell's specific gravity must be measured to confirm the state of charge. Following a minor battery recharge (such as equalizing charge that does not follow a deep discharge) specific gravity gradients are not significant, and confirming measurements may be made in less than 7 days.

The SR 4.6.3 testing of the emergency lighting is scheduled every 18 months and is subject to review and modification if experience demonstrates a more effective test schedule.

**REFERENCE**

FSAR, Section 8

## 4.7 REACTOR CONTROL ROD SYSTEM TESTS

### 4.7.1 Control Rod Drive System Functional Tests

Applicability

Applies to the surveillance of the control rod system.

Objective

To assure operability of the control rod system.

Specification

4.7.1.1 The control rod trip insertion time shall be measured for each control rod at either full flow or no flow conditions following each refueling outage prior to return to power. The maximum control rod trip insertion time for an operable control rod drive mechanism, except for the Axial Power Shaping Rods (APSRs), from the fully withdrawn position to 3/4 insertion (104 inches travel) shall not exceed 1.66 seconds at reactor coolant full flow conditions or 1.20 seconds for no flow conditions. For the APSRs it shall be demonstrated that loss of power will not cause rod movement. If the trip insertion time above is not met, the rod shall be declared inoperable.

4.7.1.2 If a control rod is misaligned with its group average by more than an indicated nine (9) inches, the rod shall be declared inoperable and the limits of Specification 3.5.2.2 shall apply. The rod with the greatest misalignment shall be evaluated first. The position of a rod declared inoperable due to misalignment shall not be included in computing the average position of the group for determining the operability of rods with lesser misalignments.

4.7.1.3 If a control rod cannot be exercised, or if it cannot be located wit absolute or relative position indications or in or out limit lights, the rod shall be declared to be inoperable.

Bases

The control rod trip insertion time is the total elapsed time from power interruption at the control rod drive breakers until the control rod has completed 104 inches of travel from the fully withdrawn position. The specified trip time is based upon the safety analysis in FSAR, Section 14, whose calculations are based on a rod drop from fully withdrawn to 2/3 inserted. Since the most accurate position indication is obtained from the zone reference switch at the 3/4 inserted position, this position is used instead of the 2/3 inserted position for data gathering.

Each control rod drive mechanism shall be exercised by a movement approximately two (2) inches of travel every two (2) weeks. This requirement shall apply to either a partial or fully withdrawn control rod at reactor operating conditions. Exercising the drive mechanisms in this manner provides assurance of reliability of the mechanisms.

A rod is considered inoperable if it cannot be exercised, if the trip insertion time is greater than the specified allowable time, or if the rod



Basis Change of 9-15-95

deviates from its group average position by more than nine (9) inches. Conditions for operation with an inoperable rod are specified in Technical Specification 3.5.2.

REFERENCES

(1) FSAR, Section 14

4.7.2 Control Rod Program Verification
(Group Vs Core Positions)

Applicability

Applies to surveillance of the control rod systems.

Objective

To verify that the designated control rod (by core position) is operating in its programmed functional position and group (rods 1 through 12, group 1-8).

Specification

4.7.2.1 Whenever the control rod drive patch is reconnected (after test, reprogramming, or maintenance), each control rod drive mechanism shall be selected from the control room and exercised by movement of sufficient travel to verify that the proper rod has responded as shown on the unit computer printout or on the input to the computer for that rod.

4.7.2.2 Whenever power or instrumentation cables to the control rod drive assemblies atop the reactor or at the bulkhead are disconnected or removed, an independent verification check of their reconnection shall be performed.

4.7.2.3 Any rod found to be improperly programmed shall be declared inoperable until properly programmed.

Bases

Each control rod has a relative and an absolute position indicator system. One set of outputs goes to the plant computer identified by a unique number associated with only one core position. The other set of outputs goes to a programmable bank of 68 edgewise meters in the control room. In the event that a patching error is made in the patch panel or connectors in the cables leading to the control rod drive assemblies or the control room meter bank is improperly transposed upon reconnection, these errors and transpositions will be discovered by a comparative check by (1) selecting a specific rod from one group (e.g., rod 1 in regulating group 6), (2) noting the program-approved core position for this rod of the group, (3) exercising the selected rod, and (4) noting that a) the computer prints out both absolute and relative position response for the approved core position, and b) the proper meter in the control room display bank indicates both absolute and relative meter positions. This type of comparative check will not assure detection of improperly connected cables inside the reactor building. For these, (Specification 4.7.2.2) it will be necessary for a responsible person, other than the one doing the work, to verify by appropriate means that each cable has been matched to the proper control rod drive assembly.



effective February 25, 1991

Correction letter of 2-4-91

4.8 EMERGENCY FEEDWATER PUMP TESTING

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 Each EFW train shall be demonstrated operable:

a) By verifying on a STAGGERED TEST BASIS:

1. at least once per 31 days or within 24 hours after reaching the Hot Shutdown condition following a plant heatup and prior to criticality, that the turbine-driven pump starts, operates for a minimum of 5 minutes and develops a discharge pressure of ≥ 1200 psig at a flow of ≥ 500 gpm through the test loop flow path.

2. at least once per 31 days by verifying that the motor driven EFW pump starts, operates for a minimum of 5 minutes and develops a discharge pressure of ≥ 1200 psig at a flow of ≥ 500 gpm through the test loop flow path.

b) At least once per 31 days by verifying that each valve (manual, power operated or automatic) in each EFW flowpath that is not locked, sealed, or otherwise secured in position, is in its correct position.

c) Prior to relying upon any steam generator for heat removal whenever the plant has been in CSD or less for > 30 days, verify proper alignment of each manual valve in each required EFW flow path, which if mispositioned may degrade EFW operation, from the 'Q' condensate storage tank to each steam generator.

d) At least once per 92 days by cycling each motor-operated valve in each flowpath through at least one complete cycle.

e) At least once per 18 months by functionally testing each EFW train and:

1. Verifying that each automatic valve in each flowpath actuates automatically to its correct position on receipt of an actual or simulated actuation signal.

2. Verifying that the automatic steam supply valves associated with the steam turbine driven EFW pump actuate to their correct positions upon receipt of an actual or simulated actuation signal. This test is not required to be performed until 24 hours after reaching 800 psig in the steam generators.

3. Verifying that the motor-driven EFW pump starts automatically upon receipt of an actual or simulated actuation signal.

4. Verifying that feedwater is delivered to each steam generator using the electric motor-driven EFW pump.

5. Verifying that the EFW system can be operated manually by over-riding automatic signals to the EFW valves.

## Bases

The monthly 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 assures valve operability when called upon to function. Testing of the turbine driven EFW pump is delayed until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test at 280°F. Testing may occur at a lower steam generator pressure if operational experience shows that sufficient steam pressure to perform the test exists.

Surveillance Requirement 4.8.1.c ensures that the EFW system is properly aligned by verifying the flow paths to each steam generator prior to relying upon any steam generator for heat removal after more than 30 days in Cold Shutdown or below. Operability of the EFW flow paths must be demonstrated before sufficient core heat is generated that would require the operation of the EFW system during a subsequent shutdown. This requirement is reasonable, based on engineering judgment, in view of other administrative controls to ensure that the flow paths are operable. To further ensure EFW system alignment, flow path operability is verified following extended outages to determine no misalignment of valves has occurred. This SR ensures that the flow path from the 'Q' CST to the steam generators is properly aligned.

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.

The automatic actuation circuitry testing and calibration will be performed per Surveillance Specification 4.1, and will be sufficient to assure that this circuitry will perform its intended function when called upon.



effective March 31, 1995

## 4.9 REACTIVITY ANOMALIES

### Applicability

Applies to potential reactivity anomalies.

### Objective

To require the evaluation of reactivity anomalies of a specified magnitude occurring during the operation of the unit.

### Specification

Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the coolant shall be periodically compared with the predicted value. If the difference between the observed and predicted steady-state concentrations reaches the equivalent of one percent in reactivity, an evaluation of this abnormal occurrance will be made to determine the cause of the discrepancy.

### Bases

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burnup and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burnup and reactivity is compared with that predicted. This process of normalization should be completed after about 10 percent of the total core burnup. Thereafter, actual boron concentration can be compared with prediction, and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than 1 percent $\Delta k/k$ would be unexpected, and its occurrence would be thoroughly investigated and evaluated.

The value of 1 percent $\Delta k/k$ is considered a safe limit since a shutdown margin of at least 1 percent $\Delta k/k$ with the most reactive rod in the fully withdrawn position is always maintained.

## 4.10 CONTROL ROOM EMERGENCY VENTILATION AND AIR CONDITIONING SYSTEM SURVEILLANCE

Applicability

Applies to the surveillance of the control room emergency ventilation and air conditioning systems.

Objective

To verify an acceptable level of efficiency and operability of the control room emergency ventilation and air conditioning systems.

Specification

4.10.1 Each train of control room emergency air conditioning shall be demonstrated Operable:

a. At least once per 31 days on a staggered test basis by:

1. Starting each unit and

2. Verifying that each unit operates for at least 1 hour and maintains the control room air temperature ≤84°F D.B.

b. At least once per 18 months by verifying a system flow rate of 9900 cfm ±10%.

4.10.2 Each train of control room emergency ventilation shall be demonstrated Operable:

a. At least once per 31 days on a Staggered Test Basis by initiating, from the Control Room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

b. At least once per 18 months or 1)after any structural maintenance on the HEPA filter or charcoal adsorber housings, or 2) following significant painting, fire, or chemical release in any ventilation zone communicating with the system by:

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c, and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 2000 cfm ±10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989 when tested at 30°C and 95% relative humidity for a methyl iodide penetration of:

a. ≤2.5% for 2 inch charcoal adsorber beds, or

b. ≤0.5% for 4 inch charcoal adsorber beds.

3. Verifying a system flow rate of 2000 cfm ±10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989 when tested at 30°C and 95% relative humidity for a methyl iodide penetration of:

   1. ≤ 2.5% for 2 inch charcoal adsorber beds, or

   2. ≤ 0.5% for 4 inch charcoal adsorber beds.

d. At least once per 18 months by:

   1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 6 inches of water while operating at a flowrate of 2000 cfm ± 10%.

   2. Verifying that on a Control Room ventilation high radiation test signal, the system automatically isolates the Control Room within 10 seconds and switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.

e. After each complete or partial replacement of the HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 2000 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove ≥ 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 2000 cfm ± 10%.

Bases

The purpose of the control room emergency ventilation system is to limit the particulate and gaseous fission products to which the control area would be subjected during an accidental radioactive release in or near the Auxiliary Building. The system is designed with 100 percent capacity filter trains which consist of a prefilter, high efficiency particulate filters, charcoal adsorbers and a fan.

Since the emergency ventilation system is not normally operated, a periodic test is required to insure operability when needed. During this test the system will be inspected for such things as water, oil, or other foreign material; gasket deterioration, adhesive deterioration in the HEPA units; and unusual or excessive noise or vibration when the fan motor is running. Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop should be determined at least once per operating cycle to show system performance capability.

Bases (Continued)

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. The charcoal adsorber efficiency test procedures should allow for obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. Tests of the charcoal adsorbers with DOP aerosol shall be performed in accordance with ANSI N510 (1975) "Standard for Testing of Nuclear Air Cleaning Systems." Any HEPA filters found defective shall be replaced with filters qualified according to Regulatory Position C.3.d of Regulatory Guide 1.52. If laboratory test results are unacceptable, all charcoal adsorbents in the system shall be replaced with charcoal adsorbent qualified according to Regulatory Guide 1.52.

The operability of the control room emergency air conditioning Systems ensure that the ambient air temperature does not exceed the allowable temperature for the equipment and instrumentation cooled by this system and the Control Room will remain habitable for Operations personnel during and following all credible accident conditions.

Operation of the systems for 15 minutes every month will demonstrate operability of the emergency ventilation and emergency air conditioning systems. All dampers and other mechanical and isolation systems will be shown to be operable.

If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significant shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

4.11 PENETRATION ROOM VENTILATION SYSTEM SURVEILLANCE

Applicability

Applies to the surveillance of the penetration room ventilation system.

Objective

To verify an acceptable level of efficiency and operability of the penetration room ventilation system.

Specification

4.11.1 At intervals not to exceed 18 months, the pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 6 inches of water at system design flow rate ($\pm$ 10%).

4.11.2 Initially and after any maintenance or testing that could affect the air distribution within the penetration room ventilation system, air distribution shall be demonstrated to be uniform within $\pm$20% across HEPA filters and charcoal adsorbers.

4.11.3 At intervals not to exceed 18 months, automatic initiation of the penetration room ventilation system shall be demonstrated.

4.11.4a The tests and sample analysis of Specification 3.13.1a, b, & c. shall be performed at intervals not to exceed 18 months or after every 720 hours of system operation and following significant painting, fire or chemical release in any ventilation zone communicating with the system.

b. Cold DOP testing shall also be performed after each complete or partial replacement of the HEPA filter bank or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon testing shall also be performed after each complete or partial replacement of the charcoal adsorber bank or after any structural maintenance on the system housing.

4.11.5 Each circuit shall be operated at least 1 hour every month. This test shall be considered satisfactory if control board indication verifies that all components have responded properly to the actuation signal.

Bases

The penetration room ventilation system is designed to collect and process potential reactor building penetration room leakage to minimize environmental activity levels resulting from post accident reactor building leaks. The system consists of a sealed penetration room, two redundant filter trains and two redundant fans discharging to the unit vent. The entire system is activated by a reactor building pressure engineered safety features signal and initially requires no operator action.

Since the system is not normally operated, a periodic test is required to show that the system is available for its engineered safety features function. During this test the system will be inspected for such things as water, oil, or other foreign material, gasket deterioration in the HEPA units, and unusual or excessive noise or vibration when the fan motor is running.

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop should be determined at least once per 18 months to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. The charcoal adsorber efficiency test procedures should allow for obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant and of the HEPA filter bank with DOP aerosol shall be performed in accordance with ANSI N510 (1975) "Standard for Testing of Nuclear Air Cleaning Systems." Any HEPA filters found defective shall be replaced with filters qualified according to Regulatory Position C.3.d. of Regulatory Guide 1.52. Radioactive methyl iodide removal efficiency tests shall be performed in accordance with ASTM D3803-1989. If laboratory test results are unacceptable, all charcoal adsorbents in the system shall be replaced with charcoal adsorbents qualified according to ASTM D3803-1989.

Operation of the system each month for 1 hour will demonstrate operability of the active system components and the filter and adsorber system. If significant painting, fire or chemical release occurs such that the HEPA filter or charcoal adsorber could become contaminated from the fumes, chemicals or foreign material, the same tests and sample analysis shall be performed as required for operational use. The determination of significant shall be made by the operator on duty at the time of the incident. Knowledgeable staff members should be consulted prior to making this determination.

## 4.12 HYDROGEN RECOMBINERS SURVEILLANCE

### Applicability

Applies to the surveillance of the hydrogen recombiner systems.

### Objective

To verify an acceptable level of efficiency and operability of the hydrogen recombiner systems.

### Specification

4.12.1 Each hydrogen recombiner system shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying during a recombiner system functional test that the minimum heater sheath temperature increases to greater than or equal to 700°F within 90 minutes. Upon reaching 700°F, increase the power setting to maximum power for 2 minutes and verify that the power meter reads greater than or equal to 60 KW.

b. At least once per 18 months by:

1. Performing a CHANNEL CALIBRATION of all recombiner instrumentation and control circuits,

2. Verifying through a visual examination that there is no evidence of abnormal conditions within the recombiner enclosure (i.e., loose wiring or structural connections, deposits of foreign materials, etc.), and

3. Verifying the integrity of the heater electrical circuits by performing a resistance to ground test following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.

4.12.2 Hydrogen concentration instruments shall be calibrated once every 18 months with proper consideration to moisture effect.

### Bases

The OPERABILITY of the recombiners for the control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water, and 3) corrosion of metals within containment. The hydrogen recombiner systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following LOCA", Rev. 2, November, 1978.

4.13 EMERGENCY COOLING POND

Applicability

Applies to the emergency cooling pond.

Objective

To verify the availability of a sufficient supply of cooling water inventory in the emergency cooling pond.

Specification

4.13.1 The emergency cooling pond shall be determined operable:

1. At least once per 24 hours by verifying the pond's indicated water level is ≥ 5 feet.

2. At least once per 24 hours during the period from June 1 through September 30 by verifying that the pond's average water temperature at the point of discharge from the pond is within its limit.

3. At least once per 12 months by making soundings of the pond and verifying an average depth of 5 feet and that the contained water volume of the pond is within its limit.

4. At least once per 12 months by a visual inspection of the loose stone (riprap) placed on the banks of the pond and of the concrete slab spillway and verifying that the earth portions of the stone covered embankments and the spillway:

   1. Have not been eroded or undercut by wave action, and

   2. Do not show apparent changes in visual appearance or other abnormal degradation from their as built condition.

Bases

The requirements of Specification 4.13 provide for verification of a sufficient water inventory in the emergency cooling pond to handle a DBA with a concurrent failure of the Dardanelle Reservoir. This specification ensures that Specification 3.11.1 is met. Monitoring temperature only during the period June 1 through September 30 of each year ensures that, during the hot summer months, the pond temperature limit is not exceeded. During other periods of the year the pond temperature will not have the potential to reach the temperature limit. Soundings are performed to ensure the water volume is within limits and that the indicated level is indicative of an equivalent water volume for accident mitigation. The measured ECP temperature at the discharge from the pond is considered a conservative average of total pond conditions since solar gain, wind speed, and thermal current effects throughout the pond will essentially be at equilibrium conditions under initial stagnant conditions. Visual inspections are performed to ensure any physical degradation is within acceptable limits to enable the ECP to fulfill its safety function. An engineering evaluation shall be performed by a qualified engineer of any apparent changes in visual appearance or other abnormal degradation to determine operability.

## 4.14 RADIOACTIVE MATERIALS SOURCES SURVEILLANCE

Applicability

Applies to leakage testing of byproduct, source, and special nuclear radioactive material sources.

Objective

To assure that leakage from byproduct, source, and special nuclear radioactive material sources does not exceed allowable limits.

Specification

Test for leakage and/or contamination shall be performed by the licensee or by other persons specifically authorized by the Commission or an agreement State, as follows:

1. Each sealed source, except startup sources subject to core flux, containing radioactive material, other than Hydrogen 3, with a half-life greater than 30 days and in any form other than gas shall be tested for leakage and/or contamination at intervals not to exceed six months.

2. The periodic leak test required does not apply to sealed sources that are stored and not being used. The sources excepted from this test shall be tested for leakage prior to any use or transfer to another user unless they have been leak tested within six months prior to the date of use or transfer. In the absence of a certificate from a transferrer indicating that a test has been made within six months prior to the transfer, sealed sources shall not be put into use until tested.

3. Each sealed startup source shall be leak tested within 31 days prior to being subjected to core flux and following repair or maintenance to the source.

4. The periodic leak test does not apply to the boronometer source. This source shall be tested for leakage at least once per 18 months.



effective: August 5, 1991

## 4.15 AUGMENTED INSERVICE INSPECTION PROGRAM FOR HIGH ENERGY LINES OUTSIDE OF CONTAINMENT

### Applicability

Applies to welds in piping systems located outside of containment where protection from the consequences of postulated ruptures is not provided by a system of pipe whip restraints, jet impingement barriers, protective enclosures and/or other measures designed specifically to cope with such ruptures.

For Arkansas Nuclear One-Unit 1 this specification applies to six welds in the main steam and main feedwater lines identified as welds 6, 7, 23, 24, 55 and 56 on Figures A-7, A-8 and A-15 of the Final Safety Analysis Report.

### Objective

To provide assurance of the continued integrity of the piping systems over their service lifetime.

### Specifications

4.15.1 At the first refueling outage period, a volumetric examination shall be performed with 100 percent inspection of each weld in accordance with the requirements of ASME Code Section XI, Inservice Inspection of Nuclear Power Plant Components, to establish system integrity and baseline data.

4.15.2 The inservice inspection at each weld shall be performed in accordance with the requirements of ASME Code Section XI, Inservice Inspection of Nuclear Power Plant Components, with the following schedule:

(The inspection intervals identified below sequentially follow the baseline examination of 4.15.1).

First Inspection Interval

| | |
|---|---|
| a. First 3-1/3 years (or nearest refueling outage) | 100% volumetric inspection of each weld |
| b. Second 3-1/3 years (or nearest refueling outage) | 100% volumetric inspection of each weld |
| c. Third 3-1/3 years (or nearest refueling outage) | 100% volumetric inspection of each weld |

Successive Inspection Intervals

| | |
|---|---|
| Every 10 years thereafter (or nearest refueling outage) | Volumetric inspection of two of the welds at the expiration of each 1/3 of the inspection interval with a cumulative 100% coverage of all welds. |

Note - The welds selected during each inspection period shall be distributed among the total number to be examined to provide a representative sampling of the conditions of the welds.

4.15.3 In the event repairs of any welds are required following any examination during successive inspection intervals, the inspection schedule for the repaired welds will revert back to the first 10 year inspection program.

4.15.4 Examinations that reveal unacceptable structural defects in a weld during an inspection under 4.15.2 should be extended to require an additional inspection of another 1/3 of the welds. If further unacceptable defects are detected in the second sampling, the remainder of the welds shall be inspected.

4.15.5 Repairs, reexamination and piping pressure tests shall be conducted in accordance with Section XI of the ASME Code.

## 4.16 SHOCK SUPPRESSORS (Snubbers)

### Applicability

This technical specification applies to all shock suppressors (snubbers). The only snubbers excluded from this requirement are those installed on nonsafety-related systems and then only if their failure or failure of the system on which they are installed would have no adverse effect on any safety-related system.

### Objective

Verify an acceptable level of operability of the shock suppressors protecting the primary system and any other safety-related system or component.

### Specification

4.16.1 The following surveillance requirements apply to all applicable shock suppressors.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers may be categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 4.16-1. The visual inspection interval for each category of snubber shall be determined based upon the criteria provided in Table 4.16.1.

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired operability, and (2) attachments to the foundation or supporting structure are functional and (3) fastners for the attachment of the snubber to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of visual inspections shall be classified as INOPERABLE and may be reclassified OPERABLE for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined operable per Specifications 4.16.1.d or 4.16.1.e, as applicable. However, when the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined inoperable and cannot be determined operable via functional testing for the purpose of establishing the next visual inspection interval. All snubbers connected to a common hydraulic fluid reservoir shall be evaluated for operability if any snubber connected to that reservoir is determined to be inoperable.

d. Functional Tests

At least once each refueling shutdown a representative sample of snubbers shall be tested using the following sample plan.

At least 10% of the snubbers required by Specification 3.16.1 shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of Specification 4.16.1.e, an additional 10% of the snubbers shall be functionally tested until no more failures are found or until all snubbers have been functionally tested.

The representative samples for the functional test sample plans shall be randomly selected from the snubbers required by Specification 3.16.1 and reviewed before beginning the testing. The review shall ensure as far as practical that they are representative of the various configurations, operating environments, range of sizes, and capacities. Snubbers placed in the same locations as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional testing results shall be reviewed at that time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

e. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1) Activation (restraining action) is achieved within the specified range in both tension and compression, except that inertia dependent, acceleration limiting mechanical snubbers may be tested to verify only that activation takes place in both directions of travel;

2) Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range;

3) Where required, the force required to initiate or maintain motion of the snubber is within the specified range in both direction of travel; and

4) For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

f. Functional Test Failure Analysis

An evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the operability of other snubbers irrespective of type which may be subject to the same failure mode.

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

If any snubber selected for functional testing either fails to activate or fails to move, i.e., frozen-in-place, the cause will be evaluated and, if caused by manufacturer or design deficiency, all snubbers of the same type subject to the same defect shall be evaluated in a manner to ensure their operability. This testing requirement shall be independent of the requirements stated in Specification 4.16.1.d for snubbers not meeting the functional test acceptance criteria.

g. Preservice Testing of Repaired, Replacement and New Snubbers

Preservice operability testing shall be performed on repaired, replacement or new snubbers prior to installation. Testing may be at the manufacturer's facility. The testing shall verify the functional test acceptance criteria in Specification 4.16.1.e.

In addition, a preservice inspection shall be performed on each repaired, replacement or new snubber and shall verify that:

1) There are no visible signs of damage or impaired operability as a result of storage, handling or installation;

2) The snubber load rating, location, orientation, position setting and configuration (attachments, extensions, etc.), are in accordance with design;

3) Adequate swing clearance is provided to allow snubber movement;

4) If applicable, fluid is at the recommended level and fluid is not leaking from the snubber system;

5) Structural connections such as pins, bearings, studs, fasteners and other connecting hardware such as lock nuts, tabs, wire and cotter pins are installed correctly.

h. Snubber Seal Replacement Program

The seal service life of hydraulic snubbers shall be monitored to ensure that the service life is not exceeded between surveillance inspections. The expected service life for the various seals, seal materials, and applications shall be determined and established based on engineering information and the seals shall be replaced so that the expected service life will not be exceeded during a period when the snubber is required to be operable. The seal replacements shall be documented and the documentation shall be retained in accordance with Specification 6.9.2.

TABLE 4.16-1
SNUBBER VISUAL INSPECTION INTERVAL

NUMBER OF INOPERABLE SNUBBERS

| Population per Category (Notes 1 and 2) | Column A Extend Interval (Notes 3 and 6) | Column B Repeat Interval (Notes 4 and 6) | Column C Reduce Interval (Notes 5 and 6) |
|---|---|---|---|
| 1 | 0 | 0 | 1 |
| 80 | 0 | 0 | 2 |
| 100 | 0 | 1 | 4 |
| 150 | 0 | 3 | 8 |
| 200 | 2 | 5 | 13 |
| 300 | 5 | 12 | 25 |
| 400 | 8 | 18 | 36 |
| 500 | 12 | 24 | 48 |
| 750 | 20 | 40 | 78 |
| 1000 or greater | 29 | 56 | 109 |

Note 1: The next visual inspection interval for a snubber category shall be determined based upon the previous inspection interval and the number of INOPERABLE snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, categories must be determined and documented before any inspection and that determination shall be the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population per category and the number of INOPERABLE snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, and C if that integer includes a fractional value of INOPERABLE snubbers as determined by interpolation.

TABLE 4.16-1 (Continued)
SNUBBER VISUAL INSPECTION INTERVAL

Note 3: If the number of INOPERABLE snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.

Note 4: If the number of INOPERABLE snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

Note 5: If the number of INOPERABLE snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of INOPERABLE snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of INOPERABLE snubbers found during the previous interval and the number in Column B to the difference in the numbers in Column B and C.

Note 6: Specified surveillance intervals may be adjusted plus or minus 25 percent to accommodate normal test and surveillance schedule intervals up to and including 48 months, with the exception that inspection of inaccessible snubbers may be deferred to the next shutdown when plant conditions allow five days for inspection. See Note 7 for definition of interval as applied to snubber visual inspections. The provisions of Specification 4 regarding surveillance intervals are not applicable.

Note 7: Interval as defined for the shock suppressors (snubbers) visual inspection surveillance requirements is the period of time starting when the unit went into cold shutdown for refueling, and ending when the unit goes into cold shutdown for its next scheduled refueling. This period of time is nominally considered to be an 18 month period, or a 24 month period based on the type of fuel being used. However, the period of time (interval) could be shorter or longer due to plant operating variables such as fuel life and operating performance.

## BASES

All safety related snubbers are required to be operable to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety related systems and then only if their failure, or failure of the system on which they are installed, would have no adverse effect on any safety related system.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to plant systems. Therefore, the required inspection interval varies based upon the number of INOPERABLE snubbers found during the previous inspection in proportion to the sizes of the various snubber populations or categories and the previous inspection interval as specified in NRC Generic Letter 90-09, "Alternative Requirements For Snubber Visual Inspection Intervals and Corrective Actions". Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the result of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety related component or system has been adversely affected by inoperability of the snubber. The engineering evaluation is performed to determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

If a review and evaluation of an INOPERABLE snubber is performed and documented to justify continued operation, and provided that all design criteria are met with the INOPERABLE snubber, then the INOPERABLE snubber would not need to be restored or replaced.

4.17 FUEL HANDLING AREA VENTILATION SYSTEM SURVEILLANCE

Applicability

Applies to the surveillance of the fuel handling area ventilation system.

Objective

To verify an acceptable level of efficiency and operability of the fuel handling area ventialtion system.

Specification

4.17.1 At intervals not to exceed 18 months, pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 6 inches of water at system design flow rate ($\pm$ 10%).

4.17.2 Initially and after any maintenance or testing that could affect the air distribution within the fuel handling area ventilation system, air distribution shall be demonstrated to be uniform within $\pm$20% across HEPA filters and charcoal adsorbers.

4.17.3.a. The tests and sample analysis of Specification 3.15.1.a,b, & c. shall be performed within 720 system operating hours prior to irradiated fuel handling operations in the auxiliary building, and prior to irradiated fuel handling in the auxiliary building following significant painting, fire or chemical release in any ventilation zone communicating with the system.

b. Cold DOP testing shall also be performed prior to irradiated fuel handling in the auxiliary building after each complete or partial replacement of a HEPA filter bank or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon testing shall also be performed prior to irradiated fuel handling in the auxiliary building after each complete or partial replacement of a charcoal adsorber bank or after any structural maintenance on the system housing.

4.17.4 The system shall be operated for at least 10 hours prior to initiation of irradiated fuel handling operations in the auxiliary building if it has not been operated for at least 10 hours within the previous 30 days.

Bases

Since the fuel handling area ventilation system may be in operation when fuel is stored in the pool but not being handled, its operability must be verified before handling of irradiated fuel. Operation of the system for 10 hours before irradiated fuel handling operations and performance of Specification 4.17.3 will demonstrate operability of the active system components and the filter and adsorber systems.

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop and air distribution should be determined once every 18 months to show system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. The charcoal adsorber efficiency test procedures should allow for obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant and of the HEPA filter bank with DOP aerosol shall be performed in accordance with ANSI N510 (1975) "Standard for Testing of Nuclear Air Cleaning Systems." Any HEPA filters found defective shall be replaced with filters qualified according to Regulatory Position C.3.d. of Regulatory Guide 1.52. Radioactive methyl iodide removal efficiency tests shall be performed in accordance with ASTM D3803-1989. If laboratory test results are unacceptable, all charcoal adsorbents in the system shall be replaced with charcoal adsorbents qualified according to ASTM D3803-1989.

## 4.18 STEAM GENERATOR TUBING SURVEILLANCE

Applicability

Applies to the surveillance of tubing of each steam generator.

Objective

To ensure integrity of the steam generator tubing through a defined inservice surveillance program, and to minimize exposure of personnel to radiation during performance of the surveillance program.

Specification

### 4.18.1 Baseline Inspection

The first steam generator tubing inspection performed according to Specifications 4.18.2 and 4.18.3.a shall be considered as constituting the baseline condition for subsequent inspections.

### 4.18.2 Examination Methods

a. Inservice inspection of steam generator tubing shall include non-destructive examination by eddy-current testing or other equivalent techniques. The inspection equipment shall provide a sensitivity that will detect defects with a penetration of 20 percent or more of the minimum allowable as-manufactured tube wall thickness except for a sleeved tube at the lower sleeve end.

b. For examination of the sleeved steam generator tubing at the lower sleeve end, the indications will be compared to those obtained during the baseline sleeved tube inspection. Significant deviations between these indications will be considered sufficient evidence to warrant designation as a degraded tube. Direct quantification of the 40 percent through-wall plugging limit is available with eddy-current testing.

### 4.18.3 Selection and Testing

The steam generator sample size is specified in Table 4.18-1. The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.18-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.18.4 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.18.5. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in both steam generators; the tubes selected for these inspections shall be selected on a random basis except:



effective August 24, 1990

a. The first sample inspection during each inservice inspection (subsequent to the baseline inspection) of each steam generator shall include:

1. All nonplugged tubes that previously had detectable wall penetrations (>20%), except tubes in which the wall penetration has been spanned by a sleeve, and

2. At least 50% of the tubes inspected shall be in those areas where experience has indicated potential problems, except where specific groups are inspected per Specification 4.18.3.a.3.

A tube inspection (pursuant to Specification 4.18.5.a.9) 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.

3. Tubes in the following groups may be excluded from the first random sample if all tubes in a group in both steam generators are inspected. The inspection may be concentrated on those portions of the tubes where imperfections were previously found. No credit will be taken for these tubes in meeting minimum sample size requirements. Where only a portion of the tube is inspected, the remainder of the tube will be subjected to the random inspection.

(1) Group A-1: Tubes within one, two or three rows of the open inspection lane.

(2) Group A-2: Unplugged tubes with sleeves installed.

(3) Group A-3: Tubes in the wedge-shaped group on either side of the lane region (Group A-1) as defined by Figure 4.18.1.

4. Tubes with axially-oriented tube end cracks (TEC) which have been left inservice for the previous cycle shall be inspected with a rotating coil eddy current technique in the area of the TEC and characterized in accordance with topical report BAW-2346P, Rev.0, during all subsequent SG inspection intervals pursuant to 4.18.4. The results of this examination may be excluded from the first random sample. Tubes with axial TECs identified during previous inspections which meet the criteria to remain in service will not be included when calculating the inspection category of the OTSG.

5. Implementation of the upper tubesheet ODIGA alternate repair criteria requires a 100% bobbin coil inspection of the non-plugged and non-sleeved tubes, spanning the defined region of the upper tubesheet, during all subsequent SG inspection intervals pursuant to 4.18.4. Tubes with ODIGA identified during previous inspections which meet the criteria to remain in service will not be included when calculating the inspection category for the OTSG. The defined region begins one inch above the upper tubesheet secondary face and ends at the nearest tube roll transition. ODIGA indications detected by the bobbin coil probe shall be characterized using rotating coil probes in accordance with ANO Engineering Report No. 00-R-1005-01.

b. All tubes which have been repaired using the reroll process will have the new roll area inspected during the inservice inspection.

c. The second and third sample inspections during each inservice inspection as required by Table 4.18-2 may be less than a full tube inspection by concentrating the inspection on those areas of the tube sheet array and on those portions of the 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. |

NOTES:

(1) In all inspections, previously degraded tubes whose degradations have not been spanned by a sleeve must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

(2) Where special inspections are performed pursuant to 4.18.3.a.3, defective or degraded tubes found as a result of the inspection shall be included in determining the Inspection Results Category for that special inspection but need not be included in determining the Inspection Results Category for the general steam generator inspection.

(3) Where special inspections are performed pursuant to 4.18.3.b, defective or degraded tube indications found in the new roll area as a result of the inspection and any indications found above the new roll area, are not included in the determination for the inspection results category of a general steam generator inspection.



-shall be implemented prior to startup from the Unit 1 Cycle 15 Refueling outage

4.18.4 Inspection Intervals

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

a. The baseline inspection shall be performed during the first refueling shutdown. Subsequent inservice inspections shall be performed at intervals of not less than 10 nor more than 24 calendar months after the previous inspection. If the results of two consecutive inspections for a given group* of tubes following service under all volatile treatment (AVT) conditions fall 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 for that group may be extended to a maximum of 40 months.

b. If the results of the inservice inspection of a steam generator performed in accordance with Table 4.18-2 at 40-month intervals for a given group* of tubes fall in Category C-3, subsequent inservice inspections shall be performed at intervals of not less than 10 nor more than 20 calendar months after the previous inspection. The increase in inspection frequency shall apply until a subsequent inspection meets the conditions specified in 4.18.4.a and the interval can be extended to 40 months.

c. Additional unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.18-2 during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary leakage in excess of the limits of Specification 3.1.6.3.b (Inservice inspection not required if leaks originate from tube-to-tubesheet welds). If the leaking tube is from either Group A-1 or A-3 as defined in Specification 4.18.3.a.3, all of the tubes in the affected group in this steam generator may be inspected in lieu of the first sample inspection specified in Table 4.18-2. If the degradation mechanism which caused the leak is limited to a specific portion of the tube length, the inspection per this paragraph may be limited to the affected portion of the tube length. If the results of this inspection fall into the C-3 category, all of the tubes in the same group in the other steam generator will also be similarly inspected.

      If the leaking tube has been repaired by the reroll process and is leaking in the new roll area, all of the tubes in the steam generator that have been repaired by the reroll process will have the new roll area inspected. If the results of this inspection fall into the C-3 category, all of the tubes with rerolled areas in the other steam generator will also be similarly inspected. This inspection will be in lieu of the first sample inspection specified in Table 4.18-2.

   2. A seismic occurrence greater than the Operating Basis Earthquake,

   3. A loss-of-coolant accident requiring actuation of the engineered safeguards, or

   4. A main steam line or feedwater line break.

---

*A group of tubes means: (a) All tubes inspected pursuant to 4.18.3.a.3, or
(b) All tubes in a steam generator less those inspected pursuant to 4.18.3.a.3.



-shall be implemented prior to startup from the Unit 1 Cycle 15 refueling outage

4.18.5 Acceptance Criteria

a. As used in this specification:

1. Tubing or Tube means that portion of the tube or sleeve which forms the primary system to secondary system pressure boundary.

2. 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.

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

4. Degraded Tube means a tube containing imperfections ≥ 20% of the nominal wall thickness caused by degradation, except where all degradation has been spanned by the installation of a sleeve or repaired by a rerolled joint.

   The reroll repair process can be used to repair tubes with defects in the upper and lower tubesheet areas as described in topical report, BAW-2303P, Revision 4.

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

6. Defect means an imperfection of such severity that it exceeds the plugging limit except where the imperfection has been spanned by the installation of a sleeve. A tube containing a defect in its pressure boundary is defective.

7. Plugging Limit means the imperfection depth at or beyond 40% of the nominal tube wall thickness for which the tube shall be sleeved, rerolled, or removed from service because it may become unserviceable prior to the next inspection. This does not apply to ODIGA indications within the defined region of the upper tubesheet. These indications shall be assessed for continued plant operation in accordance with ANO Engineering Report No. 00-R-1005-01, Rev. 1.

   Axially-oriented TEC indications in the tube that do not extend beyond the adjacent cladding portion of the tube sheet into the carbon steel portion are not included in this definition. These indications shall be assessed for continued plant operation in accordance with topical report BAW-2346P, Rev. 0.

8. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in Specification 4.18.4.c.

9. Tube Inspection means an inspection of the steam generator tube from the point of entry completely to the point of exit. For tubes that have been repaired by the reroll process within the tubesheets, that portion of the tube outboard of the new roll can be excluded from future periodic inspection requirements because it is no longer part of the pressure boundary once the repair roll is installed.

b. The steam generator shall be determined operable after completing the corresponding actions (plug, reroll, or sleeve all tubes exceeding the plugging limit and all tubes containing non-TEC through-wall cracks) required by Table 4.18-2.

4.18.6 Reports

Following each inservice inspection of steam generator tubes, the complete results of the inspection shall be reported to the NRC. This report, to be submitted within 90 days of inspection completion, shall include:

a. Number and extent of tubes inspected;

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

c. Identification of tubes plugged and tubes sleeved;

d. Number of tubes repaired by rerolling and number of indications detected in the new roll area of the repaired tubes;

e. Summary of the condition monitoring and operational assessment results when applying TEC alternate repair criteria; and

f. Summary of the condition monitoring and the operational assessment results (including growth) when applying the upper tubesheet ODIGA alternate repair criteria.

This report shall be in addition to a Special Report (per Specification 6.12.5.d) required for the results of steam generator tube inspections which fall into Category C-3 as denoted in Table 4.18-2. The Commission shall be notified of the results of steam generator tube inspections which fall into Category C-3 prior to resumption of plant operation. The written Special Report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

Bases

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. Inservice 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.

In general, steam generator tubes that are degraded beyond the repair limit can either be plugged, sleeved, or rerolled. The steam generator (SG) tubes that are plugged are removed from service by the installation of plugs at both ends of the associated tube and thus completely removing the tube from service. When the tube end cracking (TEC) alternate repair criteria is applied, axially-oriented indications found not to extend from the tube sheet cladding region into the carbon steel region may be left in service under the guidelines of topical report BAW-2346P, Rev. 0. When the upper tubesheet outer diameter intergranular attack (ODIGA) alternate repair criteria is applied, indications found within the defined region of the upper tubesheet may be left in service under the guidelines of ANO Engineering Report No. 00-R-1005-01, Rev. 1. The defined region begins one inch above the upper tubesheet secondary face and ends at the nearest tube roll transition. Following a SG inspection, an operational assessment is performed to ensure primary-to-secondary leak rates will be maintained within the assumptions of the accident analysis.

Degraded steam generator tubes can also be repaired by the installation of sleeves which span the area of degradation and serve as a replacement pressure boundary for the degraded portion of the tube, thus permitting the tube to remain in service.

Degraded steam generator tubes can also be repaired by the rerolling of the tube in the upper or lower tubesheet to create a new roll area and pressure boundary for the tube. The portion of the tube that is outboard of the repair roll is the portion of the tube closest to the primary side of the tubesheet and includes tubing from the tube end up to and including the heel expansion transition. The 1-inch repair roll is considered to be within the pressure boundary. If more than one repair roll is present, the outboard portion includes tubing from the tube end to the heel transition and the beginning of the 1-inch repair roll that is farthest from the primary side of the tubesheet. The rerolling methodology establishes a new pressure boundary inboard of the degradation, thus permitting the tube to remain in service. The degraded tube outboard of the new roll area can be excluded from future periodic inspection requirements because it is no longer part of the pressure boundary once the repair roll is installed in the tubesheet. The rerolling repair process will be used to repair defects in the upper and lower tubesheets in accordance with BAW-2303P, Revision 4.

All tubes which have been repaired using the reroll process will have the new roll area inspected during future inservice inspections. Defective or degraded tube indications found in the new roll and any indications found in the original roll need not be included in determining the Inspection Results Category for the generator inspection.

The reroll repair process can be used to repair tubes with defects in the upper and lower tubesheet areas. Installation of multiple repair rolls in a single tube is acceptable. The new roll area must be free of detectable degradation in order for the repair to be considered acceptable. After the new roll area is initially deemed acceptable, future degradation in the new roll area will be analyzed to determine if the tube is defective and needs to be removed from service or repaired. The reroll repair process is described in the topical report, BAW-2303P, Revision 4.

## TABLE 4.18-1

### MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

| | |
|---|---|
| Preservice Inspection | No |
| No. of Steam Generators per Unit | Two |
| First Inservice Inspection | Two |
| Second & Subsequent Inservice Inspection | One[1] |

Table Notation:

1 The inservice inspection may be limited to one steam generator on alternating schedule encompassing 3N% of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

## TABLE 4.18-2

### STEAM GENERATOR TUBE INSPECTION [2,3]

| 1ST SAMPLE INSPECTION | | | 2ND SAMPLE INSPECTION | | 3RD SAMPLE INSPECTION | |
|---|---|---|---|---|---|---|
| Sample Size | Result | Action Required | Result | Action Required | Result | Action Required |
| A minimum of S Tubes per S.G.[1] | C-1 | None | N/A | N/A | N/A | N/A |
| | C-2 | Plug, reroll, or sleeve defective tubes and inspect additional 2S tubes in this S.G. | C-1 | None | N/A | N/A |
| | | | C-2 | Plug, reroll, or sleeve defective tubes and inspect additional 4S tubes in this S.G. | C-1 | None |
| | | | | | C-2 | Plug, reroll, or sleeve defective tubes |
| | | | | | C-3 | Perform action for C-3 result of first sample |
| | | | C-3 | Perform action for C-3 result of first sample | N/A | N/A |
| | C-3 | Inspect all tubes in this S.G. plug, reroll, or sleeve defective tubes and inspect 2S tubes in other S.G.<br><br>Special Report to NRC pursuant to 6.12.5.d | Other S.G. is C-1 | None | N/A | N/A |
| | | | Other S.G. is C-2 | Perform action for C-2 result of second sample | N/A | N/A |
| | | | Other S.G. is C-3 | Inspect all tubes in each S.G. and plug, reroll, or sleeve defective tubes. Special Report to NRC pursuant to 6.12.5.d | N/A | N/A |

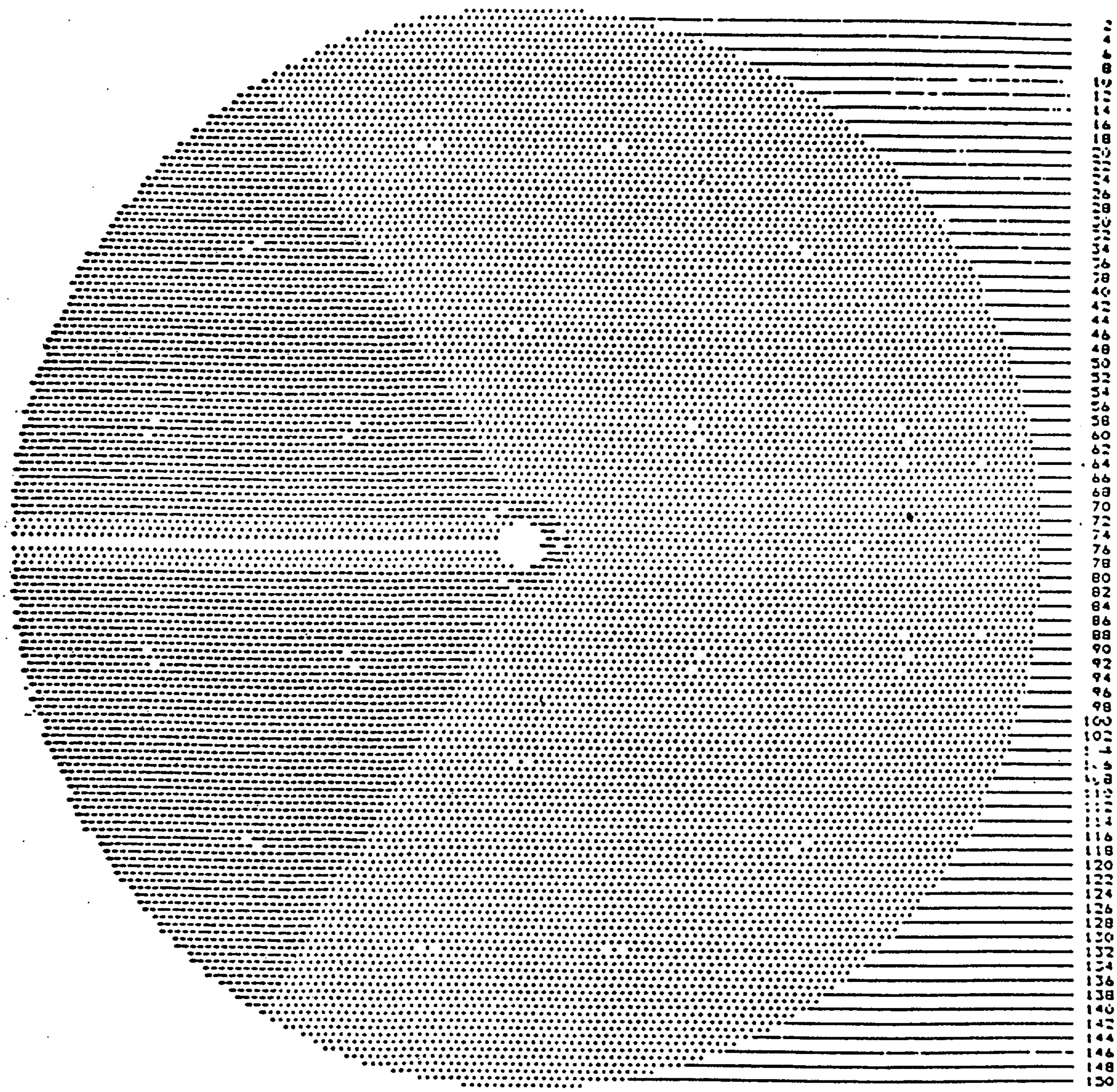
NOTES: [1] $S = 3\frac{N}{n}\%$ Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

[2] For tubes inspected pursuant to 4.18.3.a.3: No action is required for C-1 results. For C-2 results in one or both steam generators plug, reroll, or sleeve defective tubes. For C-3 results in one or both steam generators, plug, reroll, or sleeve defective tubes and provide a Special Report to NRC pursuant to 6.12.5.d.

[3] No more than ten thousand (10,000) sleeves may be installed in both ANO-1 steam generators combined.

FIGURE 4.18.1

Upper Tube Sheet View of Wedge Shaped Group (Group A-3) per Specification 4.18.3.a.3



| Description | Tube Count |
|---|---|
| Group A-1: Lane region tubes as defined in 4.18.3.a.3.(1) | 382 |
| Group A-3: Wedge shaped group depicted by darkened region of figure | 4880 |

## 4.19 FIRE DETECTION INSTRUMENTATION

DELETED

## 4.20 Fire Suppression Water System

DELETED

DELETED

Bases

DELETED

4.21 SPRINKLER SYSTEMS

DELETED

4.22 Control Room and Auxiliary Control Room Halon Systems

DELETED

## 4.23 Fire Hose Stations

DELETED

4.24 Fire Barriers

DELETED

4.25 Reactor Building Purge Filtration System

Applicability

Applies to the surveillance of the reactor building purge filtration system.

Objective

To verify an acceptable level of efficiency and operability of the reactor building purge filtration system.

Specification

4.25.1 The pressure drop across the combined HEPA filters and charcoal adsorber banks shall be demonstrated to be less than 6 inches of water at system design flow rate (±10%) within 720 system operating hours prior to initial irradiated fuel handling operations.

4.25.2 Initially and after any maintenance or testing that could affect the air distribution within the reactor building purge system, air distribution shall be demonstrated to be uniform within ±20% across HEPA filters and charcoal adsorbers.

4.25.3a. The tests and sample analysis of Specification 3.22.1.a, b, & c. shall be performed within 720 system operating hours prior to initial irradiated fuel handling operations in the reactor building, and prior to irradiated fuel handling in the reactor building following significant painting, fire or chemical release in any ventilation zone communicating with the system.

b. Cold DOP testing shall also be performed prior to irradiated fuel handling in the reactor building after each complete or partial replacement of a HEPA filter bank or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon testing shall also be performed prior to irradiated fuel handling in the reactor building after each complete or partial replacement of a charcoal adsorber bank or after any structural maintenance on the system housing.

Bases

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water at the system design flow rate will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. Pressure drop should be determined at least once per refueling period to show system performance capability.



Bases Change

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. The charcoal adsorber efficiency test procedures should allow for obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. Tests of the charcoal adsorbers with halogenated hydrocarbon refrigerant and of the HEPA filter bank with DOP aerosol shall be performed in accordance with ANSI N510 (1975) "Standard for Testing of Nuclear Air Cleaning Systems." Any HEPA filters found defective shall be replaced with filters qualified according to Regulatory Position C.3.d. of Regulatory Guide 1.52. Radioactive methyl iodide removal efficiency tests shall be performed in accordance with RDT Standard M16-IT. If laboratory test results are unacceptable, all charcoal adsorbents in the system shall be replaced with charcoal adsorbents qualified according to Regulatory Guide 1.52.

## 4.26 REACTOR BUILDING PURGE VALVES

Applicability

This specification applies to the reactor building purge supply and exhaust isolation valves.

Objective

To assure reactor building integrity.

Specification

4.26.1 The reactor building purge supply and exhaust isolation valves shall be determined closed at least once per 31 days when containment integrity is required by TS 3.6.1.

4.26.2 Prior to exceeding conditions which require establishment of reactor building integrity per TS 3.6.1, the leak rate of the purge supply and exhaust isolation valves shall be verified to be within acceptable limits per TS 4.4.1, unless the test has been successfully completed within the last three months.

Bases

Determination of reactor building purge valve closure will ensure that reactor building integrity is not unintentionally breached.

As a result of Generic Issue B-20, "Containment Leakage Due to Seal Deterioration," it was concluded that excess leakage past valve resilient seals is typically caused by severe environmental conditions and/or wear due to use. Recommended leak test frequencies of three months are deemed to be adequate to detect seal degradation of resilient seals.

The three month test need not be conducted with the precision of the Type C 10CFR50, Appendix J criteria, however the test must be sufficient to detect degradation.

## 4.27 DECAY HEAT REMOVAL

### APPLICABILITY

Applies to surveillance of the decay heat removal system and to the reactor coolant loops and associated reactor coolant pumps as needed for decay heat removal.

### OBJECTIVE

To assure the operability of the decay heat removal system and the reactor coolant loops as needed for decay heat removal.

### SPECIFICATION

4.27.1 The required reactor coolant pumps shall be determined operable once per seven (7) days by verifying correct breaker alignments and indicated power availability.

4.27.2 The required decay heat removal loop(s) shall be determined operable per Specification 4.2.2.

4.27.3 The required steam generator(s) shall be determined operable by verifying the secondary side water level to be ≥ 20 inches on the startup range at least once per 12 hours.

4.27.4 The required reactor coolant loop(s) shall be determined operable by verifying the required loop(s) to be in operation and circulating reactor coolant at least once per 12 hours.

4.27.5 The required decay heat removal loop shall be determined to be in operation at least once per 12 hours.



MAY 20 1981

## 4.28 EXPLOSIVE GAS MIXTURE

Applicability

Applies to the Waste Gas System hydrogen/oxygen analyzers.

Objective

To prevent accumulation of explosive mixtures in the waste gas system.

Specification

4.28.1 The concentration of hydrogen/oxygen in the waste gas system shall be monitored continuously by either the primary or redundant waste gas analyzer during waste gas compressing operations to the waste gas decay tanks.

4.28.2 During waste gas system operation, with no $H_2/O_2$ analyzer in service, without delay suspend all additions of waste gas to the decay tanks or take grab samples for analysis every 4 hours during degassing operations, daily during other operations. The analysis of these samples shall be completed within 8 hours of taking the sample.

Bases

This specification is to assure that the hydrogen/oxygen concentration will be kept within the limits in Figure 3.24-1 and therefore not enter the flammable region concentrations in the waste gas decay tanks.

Grab samples are to be taken every 4 hours during degassing operations when both hydrogen/oxygen analyzers are out of service. These samples are to be analyzed within 8 hours to assure that the hydrogen/oxygen concentration is within the limits in Figure 3.24-1. During other Waste Gas compressor operations, the hydrogen/oxygen concentration is not as subject to change, therefore grab samples are to be taken every 24 hours.

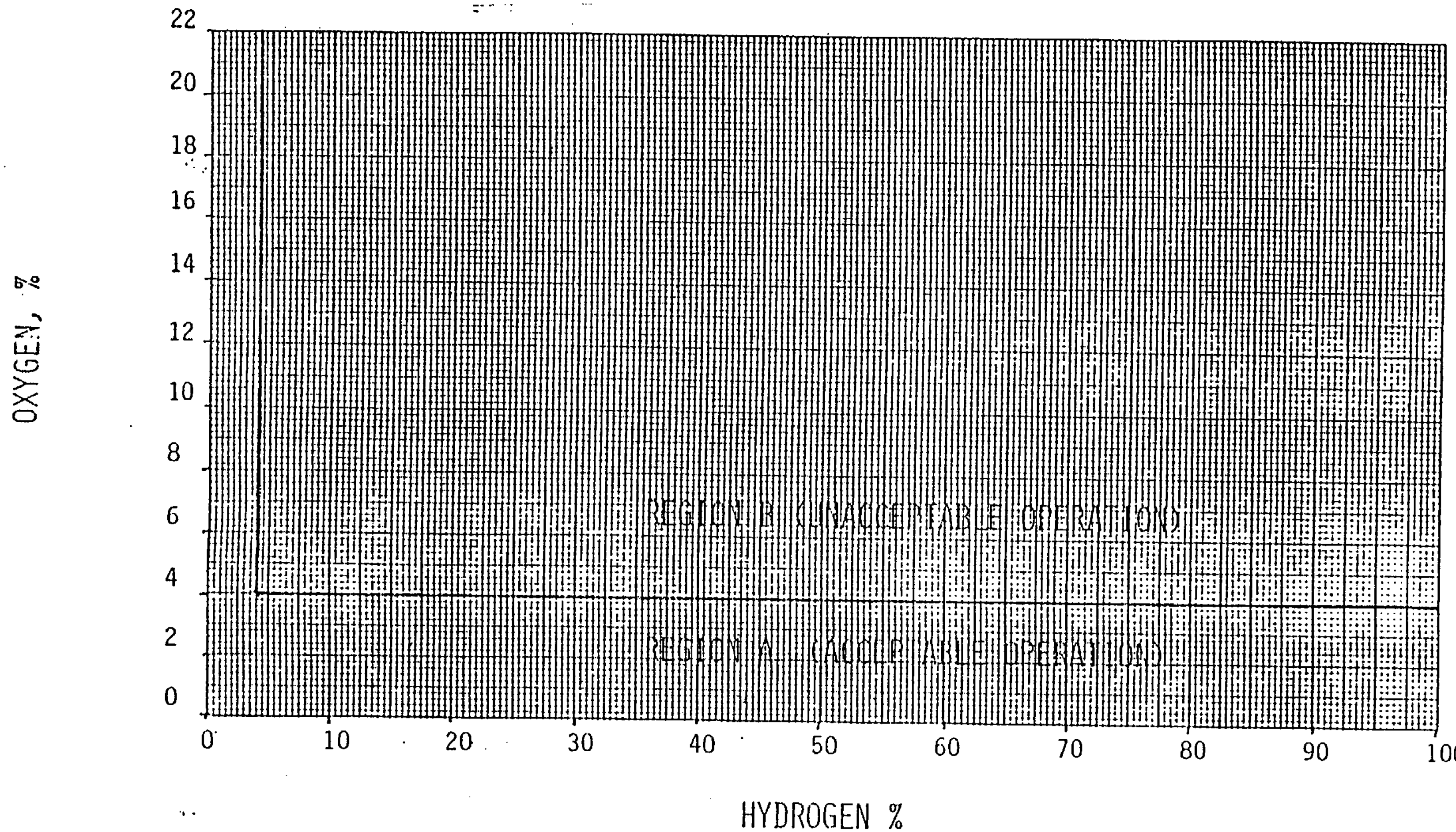



FIGURE 3.2 4-1

HYDROGEN - OXYGEN LIMITS FOR ANO-1

WASTE GAS SYSTEM

4.29 RADIOACTIVE EFFLUENTS

4.29.1 Radioactive Liquid Holdup Tanks

Applicability: At all times

Objective: To ensure that the limits of 10 CFR 20 are not exceeded.

Specification:

4.29.1 The quantity of radioactive material contained in an outside temporary radioactive liquid storage tank shall be determined to be within the limit of Specification 3.25.1 by analyzing a representative sample of the contents of the tank at least once per 7 days when radioactive materials are being added to the tank.

Bases:

This specification is provided to ensure that in the event of an uncontrolled release of the contents of the tank the resulting concentrations would be less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in the unrestricted area.

4.29.2 Radioactive Gas Storage Tanks

Applicability: At all times

Objective: To ensure meeting the requirements of Specification 3.25.2.

Specification:

4.29.2 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the limits of Specification 3.25.2 at least once per 24 hours when radioactive materials are being added to the tank and the reactor coolant activity exceeds the limits of Specification 3.1.4.1.b.

Bases:

This specification is provided so that the requirements of Specification 3.25.2 are met.

## 5.0 DESIGN FEATURES

Specifications for design features are intended to cover characteristics of importance to each of the physical barriers, and to the maintenance of safety margins in the design.

### 5.1 SITE

Applicability

Applies to the location and extent of the exclusion area.

Objective

To define the location and the size of the site area as pertains to safety.

Specification

Arkansas Nuclear One-Unit 1 is located on a site consisting of approximately 1100 acres which provides for 0.65 statute mile exclusion radius from the reactor building. This exclusion area includes certain portions of the bed and banks of the Dardanelle Reservoir which are owned by the Federal Government. An easement authorizes exclusion of all persons from these areas during periods of emergency. The site is approximately 6 statute miles WNW from the City of Russellville (Latitude 35°-18'-36" N, Longitude 93°-13'-53" W) in an area characterized by remoteness from population centers.

REFERENCE

FSAR, Section 2.2

## 5.2 REACTOR BUILDING

### Applicability

Applies to those design features of the reactor building relating to operational and public safety.

### Objective

To define the significant design features of the reactor building structure, reactor building isolation system, and penetration room ventilation system.

### Specification

#### 5.2.1 Reactor Building Structure

The reactor building completely encloses the reactor and the associated reactor coolant system. It is a fully continuous reinforced concrete structure in the shape of a cylinder with a shallow domed roof and a flat foundation slab. The cylindrical portion is prestressed by a post tensioning system consisting of horizontal and vertical tendons. The dome has a three-way post tensioning system. The foundation slab is conventionally reinforced with high strength reinforcing steel. The entire structure is lined with 1/4" welded steel plate to provide vapor tightness.

The internal net free volume of the reactor building is approximately 1.81 x $10^6$ cu. ft. The approximate inside dimensions are: diameter--116'; height--207'. The approximate thickness of the concrete forming the buildings are: cylindrical wall--3-3/4'; dome--3-1/4'; and the foundation slab--9'.

The concrete reactor building structure provides adequate shielding or both normal operation and accident situations. Design pressure and temperature are 59 psig and 286 F, respectively.

The reactor building is designed for an external atmospheric pressure of 3.0 psi greater than the internal pressure. This corresponds to a margin of 0.5 psi above the differential pressure that could be developed if the building is sealed with an internal temperature of 110 F and it is subsequently cooled to an internal temperature of less than 50 F. Since the building is designed for this pressure differential, vacuum breakers are not required.

The principal design basis for the structure is that it be capable of withstanding the internal pressure resulting from a loss of coolant accident, as defined in FSAR Section 14 with no loss of integrity. In this event the total energy contained in the water of the reactor coolant system is assumed to be released into the reactor building through a break in the reactor coolant piping. Subsequent pressure behavior is determined by the building volume, engineered safety features, and the combined influence of energy sources and heat sinks. [1]

5.2.2 Reactor Building Isolation System

Leakage through all fluid penetrations not serving accident-consequence-limiting systems is to be minimized by a double barrier so that no single, credible failure or malfunction of an active component can result in loss-of-isolation or intolerable leakage. The installed double barriers take the form of closed piping systems, both inside and outside the reactor building and various types of isolation valves. [2]

5.2.3 Penetration Room Ventilation System

This system is designed to collect, control, and minimize the release of radioactive material from the reactor building to the environment in post-accident conditions. It may also operate intermittently during normal conditions as required to maintain satisfactory temperature in the penetrations rooms. When the system is in operation, a slightly negative pressure will be maintained in the penetration room to assure inleakage. [3]

REFERENCES:

(1) FSAR Section 5.1

(2) FSAR Section 5.2.5

(3) FSAR Section 6.5

## 5.3 REACTOR

### Specification

5.3.1 Reactor Core

5.3.1.1 The reactor shall contain 177 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide pellets. Limited substitutions of stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

5.3.1.2 The reactor core approximates a right circular cylinder with an equivalent diameter of 128.9 inches and an active height of 144 inches. The active fuel length is approximately 142 inches.(2)

5.3.1.3 The average enrichment of the initial core is a nominal 2.62 weight percent of U-235. Three fuel enrichments are used in the initial core. The highest enrichment is less than 3.5 weight percent U-235.

5.3.1.4 There are 60 full-length control rod assemblies (CRA) and 8 axial power shaping rod assemblies (APSRA) distributed in the reactor core as shown in FSAR Figure 3-60. The full-length CRA contain a 134-inch length of silver-indium-cadminum alloy clad with stainless steel. Each APSRA contains a 63-inch length of Inconel-600 alloy.(3)

5.3.1.5 The initial core had 68 burnable poison spider assemblies with similar dimensions as the full-length control rods. The cladding is Zircaloy-4 filled with alumina-boron and placed in the core as shown in FSAR Figure 3-2.

5.3.1.6 Reload fuel shall conform to the design and evaluation described in FSAR and shall not exceed an enrichment of 4.1 weight percent of U-235.

5.3.2 Reactor Coolant System

5.3.2.1 The reactor coolant system is designed and constructed in accordance with code requirements.(4)

5.3.2.2 The reactor coolant system and any connected auxiliary systems exposed to the reactor coolant conditions of temperature and pressure, are designed for a pressure of 2500 psig and a temperature of 650 F. The pressurizer and pressurizer surge line are designed for a temperature of 670 F. (5)

5.3.2.3 The reactor coolant system volume is less than 12,200 cubic feet.

REFERENCES:


(1) FSAR, Section 3.2.1

(2) FSAR, Section 3.2.2

(3) FSAR, Section 3.2.4.2

(4) FSAR, Section 4.1.3

(5) FSAR, Section 4.1.2




Correction letter of 12-17-90

## 5.4 NEW AND SPENT FUEL STORAGE FACILITIES

### Applicability

Applies to storage facilities for new and spent fuel assemblies.

### Objective

To assure that both new and spent fuel assemblies will be stored in such a manner that an inadvertent criticality could not occur.

### Specification

#### 5.4.1 New Fuel Storage

1. New fuel assemblies may be stored in the Fresh Fuel Storage Rack (FFSR). The FFSR consists of a nine by eight array of storage cells on nominal center to center distance of 21 inches in both directions. Ten interior storage cells, as shown in Figure 5.4-1, are precluded from use and will be physically blocked prior to any storage in the fresh fuel rack. This configuration is sufficient to maintain a $K_{eff}$ of less than 0.98 with optimum moderation and 0.95 under normal conditions, based on fuel with an enrichment of 4.1 weight percent U-235.

2. New fuel may also be stored in the spent fuel pool or in its shipping containers.

#### 5.4.2 Spent Fuel Storage

1. The spent fuel racks are designed and shall be maintained so that the calculated effective multiplication factor is no greater than 0.95 (including all known uncertainties) when the pool is flooded with unborated water.

2. The spent fuel pool and the new fuel pool racks are designed as seismic Class I equipment.

### REFERENCES

FSAR, Section 9.6

FIGURE 5.4-1 ANO FFSR LOADING PATTERN

<----- NORTH

| | | | | | | | |
|---|---|---|---|---|---|---|---|
| | | | | | | | |
| | | | | | | | |
| | | NO | | | NO | | |
| | | | NO | NO | | | |
| | | | NO | NO | | | |
| | | | NO | NO | | | |
| | | NO | | | NO | | |
| | | | | | | | |
| | | | | | | | |

"NO" Indicates a location in which fuel loading is prohibited.

## 6.0 ADMINISTRATIVE CONTROLS

### 6.1 RESPONSIBILITY

6.1.1 The ANO-1 plant manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 An individual with an active Senior Reactor Operator (SRO) license shall be designated as responsible for the control room command function while the unit is above the Cold Shutdown condition. With the unit not above the Cold Shutdown condition, an individual with an active SRO license or Reactor Operator license shall be designated as responsible for the control room command function.

### 6.2 ORGANIZATION

#### 6.2.1 OFFSITE AND ONSITE ORGANIZATIONS

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements including the unit specific titles of those personnel fulfilling its responsibilities of the positions delineated in these Technical Specifications shall be documented in the Safety Analysis Report (SAR).

b. The ANO-1 plant manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. A specified corporate executive shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety. The specified corporate executive shall be documented in the SAR.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

UNIT STAFF

6.2.2 The operations manager or the assistant operations manager shall hold a senior reactor operator license. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

6.2.2.1 Administrative controls shall be established to limit the amount of overtime worked by plant staff performing safety-related functions. These administrative controls shall be in accordance with the guidance provided by the NRC Policy Statement on working hours (Generic Letter 82-12).

## 6.3. FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable position, except for the designated radiation protection manager, who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

## 6.4 DELETED

## 6.5 DELETED



AUG 26 1999

Table 6.2-1

ARKANSAS NUCLEAR ONE

MINIMUM SHIFT CREW COMPOSITION #

UNIT 1

| LICENSE CATEGORY | ABOVE COLD SHUTDOWN | COLD AND REFUELING SHUTDOWNS |
|---|---|---|
| SOL | 2 | 1* |
| OL | 2 | 1 |
| NON-LICENSED | 3 | 1 |

*Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising refueling operations after the initial fuel loading.

#Shift crew composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1.

Additional Requirements:

1. At least one licensed Operator shall be in the control room when fuel is in the reactor.

2. At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.

3. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

4. All refueling operations after the initial fuel loading shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

5. When the unit is above the Cold Shutdown condition, an individual shall provide advisory technical support for the unit operations shift supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift.

6.6 DELETED

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The facility shall be placed in at least hot shutdown within one hour.

b. The Nuclear Regulatory Commission shall be notified pursuant to 10 CFR 50.72 and a report submitted pursuant to the requirements of 10 CFR 50.36 and Specification 6.6.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November, 1972.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. (Deleted)

e. (Deleted)

f. Fire Protection Program Implementation.

g. New and spent fuel storage.

h. Offsite Dose Calculation Manual and Process Control Program implementation at the site.

6.8.2 (Deleted)

6.8.3 (Deleted)

6.8.4 The Reactor Building Leakage Rate Testing Program shall be established, implemented, and maintained:

A program shall be established to implement the leakage rate testing of the reactor building as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated reactor building internal pressure for the design basis loss of coolant accident, $P_a$, is 54 psig.

The maximum allowable reactor building leakage rate, $L_a$, shall be 0.20% of containment air weight per day at $P_a$.

Reactor building leakage rate acceptance criteria is $\leq 1.0\ L_a$. During the first unit startup following each test performed in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Reactor Building Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the Reactor Building Leakage Rate Testing Program.

6.8.5 The Radioactive Effluent Controls Program shall be established, implemented, and maintained:

This program conforms with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

a. Limitations on the functional capability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM;

b. Limitations on the concentrations of radioactive material released in liquid effluents to unrestricted areas conforming to 10 CFR 20, Appendix B, Table II, Column 2;

c. Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

d. Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas, conforming to 10 CFR 50, Appendix I;

e. Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

f. Limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I;

g. Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas beyond the site boundary conforming to the dose associated with 10 CFR 20, Appendix B, Table II, Column 1;

h. Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.



Correction letter of 10-20-98

6.9 <u>DELETED</u>

6.10 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.11 HIGH RADIATION AREA

6.11.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10CFR20, each high radiation area (as defined in 20.202(b)(3) of 10CFR20) in which the intensity of radiation is 1000 mrem/hr or less shall be barricaded and conspicuously posted as a high radiation area and shall be controlled by requiring the issuance of a radiation work permit. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a pre-set integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the radiation work permit.

6.11.2 The requirements of 6.11.1 above, shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and access to these areas shall be maintained under the administrative control of the shift supervisor on duty and/or the designated radiation protection manager.

## 6.12 REPORTING REQUIREMENTS

6.12.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the Administrator of the appropriate NRC Regional Office unless otherwise noted.

### 6.12.2 Routine Reports

#### 6.12.2.1 Startup Report

A summary report of plant startup and power escalation testing shall be submitted following: 1) receipt of an operating license, 2) amendment to the license involving a planned increase in power level, 3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and 4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

Startup reports shall be submitted within 1) 90 days following completion of the startup test program, 2) 90 days following resumption or commencement of commercial power operation, or 3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

#### 6.12.2.2 Occupational Exposure Data Report 1/

An Occupational Exposure Data Report for the previous calendar year shall be submitted prior to March 1 of each year. The report shall contain a tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man rem exposure according to work and job functions, 2/ e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling.

1/ A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

2/ This tabulation supplements the requirements of 20.407 of 10 CFR Part 20.

The dose assignments to various duty functions may be estimates based on pocket dosimeter, TLD, or film badge measurements. Small exposures totaling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

6.12.2.3 Monthly Operating Report

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis by the 15th of each month following the calendar month covered by the report.

6.12.2.4 Annual Report

All challenges to the pressurizer electromatic relief valve (ERV) and pressurizer safety valves shall be reported annually.

6.12.2.5 Annual Radiological Environmental Operating Report *

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 15 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

6.12.2.6 Radioactive Effluent Release Report **

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

* A single submittal may be made for ANO. The submittal should combine those sections that are common to both units.

** A single submittal may be made for ANO. The submittal should combine those sections that are common to both units. The submittal shall specify the releases of radioactive material from each unit.

6.12.3 CORE OPERATING LIMITS REPORT

6.12.3.1 The core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle or prior to any remaining part of a reload cycle for the following Specifications:

| | |
|---|---|
| 2.1 | Safety Limits, Reactor Core - Axial Power Imbalance protective limits and Variable Low RCS Pressure-Temperature Protective Limits |
| 2.3.1 | Reactor Protection System trip setting limits - Protection System Maximum Allowable Setpoints for Axial Power Imbalance and Variable low RC system pressure |
| 3.1.8.3 | Minimum Shutdown Margin for Low Power Physics Testing |
| 3.5.2.1 | Allowable Shutdown Margin limit during Power Operation |
| 3.5.2.2 | Allowable Shutdown Margin limit during Power Operation with inoperable control rods |
| 3.5.2.4 | Quadrant power Tilt limit |
| 3.5.2.5 | Control Rod and APSR position limits |
| 3.5.2.6 | Reactor Power Imbalance limits |

6.12.3.2 The analytical methods used to determine the core operating limits addressed by the individual Technical Specification shall be those previously reviewed and approved by the NRC in Babcock & Wilcox Topical Report BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses" (the approved revision at the time the reload analyses are performed). The approved revision number shall be identified in the CORE OPERATING LIMITS REPORT.

6.12.3.3 The core operating limits shall be determined so that all applicable limits (e.g. fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.12.3.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance for each reload cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

6.12.4 Reactor Building Inspection Report

6.12.4.1 Any degradation exceeding the acceptance criteria of the containment structure detected during the tests required by the ANO Containment Inspection Program shall undergo an engineering evaluation within 60 days of the completion of the inspection surveillance. The results of the engineering evaluation shall be reported to the NRC within an additional 30 days of the time the evaluation is completed. The report shall include the cause of the condition that does not meet the acceptance criteria, the applicability of the conditions to the other unit, the acceptability of the concrete containment without repair of the item, whether or not repair or replacement is required and, if required, the extent, method, and completion date of necessary repairs, and the extent, nature, and frequency of additional examinations.

6.12.5 Special Reports

Special reports shall be submitted to the Administrator of the appropriate Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification.

a. Deleted

b. Inoperable Containment Radiation Monitors, Specification 3.5.1, Table 3.5.1-1.

c. Deleted

d. Steam Generator Tubing Surveillance - Category C-3 Results, Specification 4.18.

e. Miscellaneous Radioactive Materials Source Leakage Tests, Specification 3.12.2.

f. Deleted

g. Deleted

h. Deleted

i. Deleted

j. Degraded Auxiliary Electrical Systems, Specification 3.7.2.H.

k. Inoperable Reactor Vessel Level Monitoring Systems, Table 3.5.1-1

l. Inoperable Hot Leg Level Measurement Systems, Table 3.5.1-1

m. Inoperable Main Steam Line Radiation Monitors, Specification 3.5.1, Table 3.5.1-1.

6.13 (DELETED)



effective April 1, 1991

## 6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

The ODCM shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and in the conduct of the radiological environmental monitoring program.

The ODCM shall also contain the radioactive effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Radiological Environmental Operating and Radioactive Effluent Release Reports required by Specifications 6.12.2.5 and 6.12.2.6.

Licensee initiated changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained. This documentation shall contain:

   1. Sufficient information to support the change(s) together with the appropriate analyses or evaluations justifying the change(s), and

   2. A determination that the change(s) maintain the levels of radioactive effluent control required by 10 CFR 20.1302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations;

b. Shall become effective after approval of the General Manager, Plant Operations; and

c. Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change in the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed and shall also indicate the date (i.e., month and year) the change was implemented.