2.0 LIMITING CONDITIONS FOR OPERATION

2.6 Containment System

Applicability

Applies to the reactor containment system.

Objective

To assure the integrity of the reactor containment system.

Specifications

(1) Containment Integrity

a. Containment integrity shall not be violated unless the reactor is in the cold shutdown condition.

b. Containment integrity shall not be violated when the reactor vessel head is removed if the boron concentration is less than refueling concentration.

c. Except for testing one CEDM at a time, positive reactivity changes shall not be made by CEA motion or boron dilution unless the containment integrity is intact.

d. Prior to the reactor going critical after a refueling outage, an administrative check will be made to confirm that all "locked closed" manual containment isolation valves are closed and locked.

e. The containment purge isolation valves will be closed unless the reactor is in a cold or refueling shutdown condition.

(2) Internal Pressure

The internal pressure shall not exceed 3 psig (except for containment leak rate tests).

Basis

The reactor coolant system conditions of cold shutdown assure that no steam will be formed and, hence, there would be no pressure buildup in the containment if the reactor coolant system ruptures. The shutdown margins are selected based on the type of activities that are being carried out. The refueling boron concentration provides a shutdown margin which precludes criticality under any circumstances. Each CEDM must be tested and some have two CEA's attached.

Regarding internal pressure limitations, the containment design pressure of 60 psig would not be exceeded if the internal pressure before a major loss-of-coolant accident were as much as 30 psig.[1] The containment integrity will be protected if the visual check of all "locked closed" manual isolation valves to verify them closed is made prior to plant start-up after an extended outage where one or more valves could inadvertently be left open. Operation of the purge isolation valves is prevented during normal operations due to the size of the valves (42 inches) and a concern about their ability to close against the differential pressure that could result from a LOCA or MSLB.

References

(1) FSAR, Section 14.16; Figure 14.16-2

TABLE 3-3 (Continued)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF MISCELLANEOUS INSTRUMENTATION AND CONTROLS

	Channel Description	Surveillance Function	Frequency	Surveillance Method
25.	Containment Purge Isolation Valves (PCV-742A,B,C,&D)	Check	M	Verify valve position using control room indication.

Q - Quarterly
S - Each Shift
D - Daily
M - Monthly
A - Annually
R - 18 Months
P - Prior to each startup if not performed within previous week.
PM - Prior to scheduled cold leg cooldown below 300°F; monthly whenever temperature remains below 300°F and reactor vessel head is installed.

conservative results.

Type B test methods may be substituted where appropriate. Each valve to be tested shall be closed by normal operation and without any preliminary exercising or adjustments (e.g., no tightening of valve after closure by valve motor).

Detailed supplementary criteria that establish specific test requirements to fulfill these Specifications are provided in reference 3 which is included as an enclosure to this Specification.

b. Acceptance Criteria

The combined leakage rate for all components subject to Type B and C tests and subject to the 0.6 L_a leakage limit shall not exceed 60 percent of L_a. For the purge isolation valve tests, the measured purge valve leakage rate shall be substituted for the purge valve leakage rate from the last complete Type B and C test and the total leak rate recomputed.

Leakage of the containment air purge isolation valves shall not exceed 18,000 standard cubic centimeters per minute (SCCM). If the leakage rate is determined to be greater than 18,000 SCCM, repairs shall be initiated immediately in order to meet this acceptance criteria.

c. Frequency

Type C tests shall be performed during each reactor shutdown for major refueling but in no case at intervals greater than two years, except that containment purge isolation valves shall be leakage tested prior to bringing the reactor out of cold or refueling shutdown and after the purge valves are closed for the last time.

(4) Specific Testing Requirements

Any major modification or replacement of components of the primary reactor containment performed after the initial preoperational leakage rate test shall be followed by either a Type A test, Type B test, or a Type C test of the area affected by the modification and shall meet the applicable acceptance criteria.

(5) Inspection and Reporting of Tests

a. Containment Inspection

A detailed visual examination of critical areas and general inspection of the accessible interior and exterior surfaces of the containment structures and components shall be performed at each reactor shutdown for a refueling outage and prior to any Type A test to uncover any evidence of structural deterioration which may affect the containment's structural integrity leaktightness. If there is evidence of significant structural deterioration, Type A tests shall not be performed until corrective action is taken in accordance with repair procedures, nondestructive examinations, and tests as specified in the construction code under which rules the containment was built. Such structural deterioration and corrective actions taken shall be reported as part of the Type A test report.

b. Report of Test Results

The initial Type A test shall be the subject of a summary technical report submitted to the Commission after the conduct of the test. This report shall include a schematic arrangement of the leakage rate measurement system, the instrumentation used, the supplemental test method, and the test program selected as applicable to the initial test, and all subsequent periodic tests. The report shall contain an analysis and interpretation of the leakage rate test data to the extent necessary to demonstrate the acceptability of the containment's leakage rate in meeting the acceptance criteria.

Periodic test leakage rate results of Type A, B, and C tests that meet the acceptance criteria shall be reported in the licensee's operating report. Leakage test results of Type A, B, and C tests that fail to meet the acceptance criteria shall be reported in a separate summary that includes an analysis and interpretation of the test data, the least-squares fit analysis, and the structural conditions of the containment or components, if any, which contributed to the failure in meeting the acceptance criteria. Results and analyses of the supplemental verification test employed to demonstrate the validity of the leakage rate test measurements shall also be included.

(6) Recirculation Heat Removal Systems

a. Testing Requirements

The portion of the shutdown cooling system that is outside the containment shall be tested either by use in normal operation or hydrostatically tested at 250 psig at the interval specified in the 3.5(3)a(iii).

leakage rate and duplicates the pre-operational leak rate test of 30 psig. The specification provides relationships for relating in a conservative manner the measured leakage of air at 30 psig to the potential leakage of a steam-air mixture at 60 psig and 288°F. The specification also allows for possible deterioration of the leakage rate between tests, by requiring that only 70% or 80% of the allowable leakage rates actually be measured. The basis for the deterioration allowances is arbitrary judgments which are believed to be conservative and which will be confirmed or denied by periodic testing. If indicated to be necessary, the deterioration allowances will be altered based on experience. The durations for the integrated leakage rate tests are established to provide a minimum level of accuracy and to allow for daily cyclic variation in temperature and thermal radiation.

The frequency of the periodic integrated leakage rate test is keyed to the refueling outage schedule for the reactor, because these tests can best be performed during refueling shutdowns. The initial core loading is designed for approximately 12 months of power operation; thus, the first refueling outage will occur approximately 18 months after initial criticality. Subsequent refueling outages are scheduled at approximately 12-month intervals, although larger periods may be utilized.

The specified frequency of periodic integrated leakage rate tests is based on three major considerations. First is the low probability of leaks in the liner because of the test of the leaktightness of the welds during erection and conformance of the complete containment to a low leak rate at 60 psig during preoperational testing, which is consistent with 0.1% leakage at design basis accident conditions and absence of any significant stresses in the liner during reactor operation. Second is the more frequent testing, at the full accident pressure, of those portions of the containment envelope that are most likely to develop leaks during reactor operation (penetrations and isolation valves) and the low value (0.60) of the total leakage that is specified as acceptable from penetrations and isolation valves. Third is the tendon stress surveillance program, which provides assurance that an important part of the structural integrity of the containment is maintained.

Integrity tests of the purge isolation valves are established to identify excessive degradation of the resilient seats of these valves. Simultaneous testing of redundant purge valves from a leak test connection accessible from outside containment provides adequate testing. The testing method is identical to the Type C purge isolation valve test performed in accordance with 10 CFR Part 50, Appendix J. For leakages found to be greater than 18,000 SCCM, repairs shall be initiated to ensure these valves meet the acceptance criteria.

3.0 SURVEILLANCE REQUIREMENTS

3.5 Containment Tests (Continued)

The limiting leakage rates from the shutdown cooling system are judgment values based primarily on assuring that the components could operate without mechanical failure for a period on the order of 200 days after a design basis accident. The test pressure (250 psig) achieved either by normal system operation or by hydrostatic testing gives an adequate margin over the highest pressure within the system after a design basis accident. Similarly, the hydrostatic test pressure for the return lines from the containment to the shutdown cooling system (100 psig) gives an adequate margin over the highest pressure within the lines after a design basis accident.

A shutdown cooling system leakage of one gpm will limit off-site exposures due to leakage to insignificant levels relative to those calculated for direct leakage from the containment in the design basis accident. The safety injection system pump rooms are equipped with individual charcoal filters which are placed into operation by means of switches in the control room. The radiation detectors in the auxiliary building exhaust duct are used to detect high radiation level. The one gpm leak rate is sufficiently high to permit prompt detection and to allow for reasonable leakage through the pump seals and valve packings, and yet small enough to be readily handled by the pumps and radioactive waste system. Leakage to the safety injection system pump room sumps will be returned to the spent regenerant tanks. Additional makeup water to the containment sump inventory can be readily accommodated via the charging pumps from either the SIRW tank or the concentrated boric acid storage tanks.

In case of failure to meet the acceptance criteria for leakage from the shutdown cooling system or the penetrations, it may be possible to effect repairs within a short time. If so, it is considered unnecessary and unjustified to shutdown the reactor. The times allowed for repairs are consistent with the times developed for other engineered safeguards components.

A reduction in prestressing force and changes in physical conditions are expected for the prestressing system. Allowances have been made in the reactor building design for the reduction and changes. The inspection results for each tendon shall be recorded on the forms provided for that purpose and comparison will be made with the previous test results and the initial quality control records. Force-time trend lines will also be established and maintained for each of the surveillance tendons.

If the force-time trend line, as extrapolated, falls below the predicted force-time curve for one or more surveillance tendons, then before the next scheduled surveillance inspection, an investigation shall be made to determine whether the rate of force reduction is indeed occurring for other tendons. If the rate of reduction is confirmed, the investigation shall be extended so as to identify the cause of the rate of force reduction. The extension of the investigation shall determine the needed changes in the surveillance inspection schedule and the criteria and initial planning for corrective action. If the force-time trend lines of the surveillance tendons at any time exceed the upper bound curve of the band on the force-time graph, an investigation shall be made to determine the cause.

DISCUSSION

As a result of the numerous reports on unsatisfactory performance on the resilient seats for the isolation valves in containment purge and vent lines (addressed in OI&E Circular 77-11, dated September 6, 1977), Generic Issue B-20, "Containment Leakage Due to Seal Deterioration", was established to evaluate this concern and establish an appropriate testing frequency for the isolation valves. For the Fort Calhoun Station, Omaha Public Power District initiated an in-depth test program which initially included leakage testing of the containment purge isolation valves monthly. After several months, this testing was reduced to a bi-monthly frequency and then quarterly. After about two years of testing, the program was discontinued since no degradation of the seals was identified.

The Commission's letters dated July 28, 1981 and February 24, 1982 recommended a test frequency of at least once every six months for passive purge systems. The July 28, 1981 letter also addressed a concern regarding the ability of the large butterfly type purge valves to properly operate under an accident condition. To resolve this concern, the District had previously committed to disabling these valves when not in a cold shutdown or refueling condition. The Omaha Public Power District herewith submits a Technical Specification change to include a purge isolation valve testing program and that incorporates our position on operability of the purge isolation valves.

The proposed leakage testing program, to be conducted following cold or refueling shutdown outages, is identical to the Type C purge isolation valve leakage test that has previously been conducted every refueling outage, or intervals not exceeding two years. Because of the success of the District's test program, as discussed above, and our commitment to leave these valves shut, the six-month test frequency recommended by the Commission has not been incorporated into the amendment application. Our experience has shown that leakage problems with the containment purge isolation valves result from valve operation. Accordingly, we have committed to a test frequency that ensures the valves will be leak tested following valve operation and prior to returning to hot plant conditions.

A leak test connection that is accessible from outside containment is used to pressurize the line between the redundant purge valves, inside and outside containment, to 60 psig. The leakage measurements are then made using appropriate instrumentation/equipment. Should the measured leakage rate exceed 18,000 standard cubic centimeters per minute (SCCM), corrective repairs will be initiated immediately to bring the leakage below 18,000 SCCM. The 18,000 SCCM leakage figure has been used as the maximum allowable leakage in the performance of Type C leakage tests and inservice inspections for these valves. This specific limit of 18,000 SCCM leakage provides a means to measure and identify excessive seal degradation and ensures that corrective measures are carried out as needed.

To ensure Technical Specification adherence, the measured purge valve leakage rate is recomputed with leakage rates for all other Type B and C penetrations, and the integrated leakage rate must be less than or equal to 0.6 L_a. If it is determined that the integrated leakage rate is greater than 0.6 L_a, repairs shall be initiated immediately. The action requirements to be followed if the leakage criteria is not met are covered by Technical Specification 2.6(1)a and 2.0.1(1). Accordingly, the existing action statement, paragraph 2 in Technical Specification 3.5(3)b, allowing 48 hours for repairs before action is required has been deleted, since it was significantly less conservative than 2.0.1(1).

In response to the Commission's February 24, 1982 letter, the District has added a Limiting Condition for Operation (LCO) to require the purge isolation valves to remain closed except when the plant is in a cold or refueling shutdown mode. The February 24, 1982 letter also recommended that the LCO include a requirement to remove power to these valves when not in a cold or refueling shutdown. However, if power is removed, there would be no valve position indication for these valves in the control room; therefore, the District believes that to better protect the safety of the public, the power to these valves should not be disabled. The recommended Commission requirement to verify valve position monthly has been incorporated into Table 3-3 of the Fort Calhoun Station Technical Specifications.

2.0 LIMITING CONDITIONS FOR OPERATION

2.6 Containment System

Applicability

Applies to the reactor containment system.

Objective

To assure the integrity of the reactor containment system.

Specifications

(1) Containment Integrity

a. Containment integrity shall not be violated unless the reactor is in the cold shutdown condition.

b. Containment integrity shall not be violated when the reactor vessel head is removed if the boron concentration is less than refueling concentration.

c. Except for testing one CEDM at a time, positive reactivity changes shall not be made by CEA motion or boron dilution unless the containment integrity is intact.

d. Prior to the reactor going critical after a refueling outage, an administrative check will be made to confirm that all "locked closed" manual containment isolation valves are closed and locked.

(2) Internal Pressure

The internal pressure shall not exceed 3 psig (except for containment leak rate tests).

Basis

The reactor coolant system conditions of cold shutdown assure that no steam will be formed and, hence, there would be no pressure buildup in the containment if the reactor coolant system ruptures. The shutdown margins are selected based on the type of activities that are being carried out. The refueling boron concentration provides a shutdown margin which precludes criticality under any circumstances. Each CEDM must be tested and some have two CEA's attached.

Regarding internal pressure limitations, the containment design pressure of 60 psig would not be exceeded if the internal pressure before a major loss-of-coolant accident were as much as 3 psig.(1) The containment integrity will be protected if the visual check of all "locked closed" manual isolation valves to verify them closed is made prior to plant start-up after an extended outage where one or more valves could inadvertently be left open.

References

(1) FSAR, Section 14.16; Figure 14.16-2

conservative results.

Type B test methods may be substituted where appropriate. Each valve to be tested shall be closed by normal operation and without any preliminary exercising or adjustments (e.g., no tightening of valve after closure by valve motor).

Detailed supplementary criteria that establish specific test requirements to fulfill these Specifications are provided in reference 3 which is included as an enclosure to this Specification.

b. Acceptance Criteria

The combined leakage rate for all components subject to Type B and C tests and subject to the 0.6 L_a leakage limit shall not exceed 60 percent of L_a.

If at any time it is determined that a leakage rate is greater than 60 percent of L_a, repairs shall be initiated immediately. If repairs are not completed and conformance to the acceptance criteria is not demonstrated within 48 hours, the reactor shall be shut down and depressurized until repairs are completed and the local leakage meets this acceptance criteria.

c. Frequency

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

(4) Specific Testing Requirements

Any major modification or replacement of components of the primary reactor containment performed after the initial preoperational leakage rate test shall be followed by either a Type A test, Type B test, or a Type C test of the area affected by the modification and shall meet the applicable acceptance criteria.

(5) Inspection and Reporting of Tests

a. Containment Inspection

A detailed visual examination of critical areas and general inspection of the accessible interior and exterior surfaces of the containment structures and components shall be performed at each reactor shutdown for a refueling outage and prior to any Type A test to uncover any evidence of structural deterioration which may affect the containment's structural integrity leaktightness. If there is evidence of significant structural deterioration, Type A tests shall not be performed until corrective action is taken in accordance with repair procedures, nondestructive examinations, and tests as specified in the construction code under which rules the containment was built. Such structural deterioration and corrective actions taken shall be reported as part of the Type A test report.

b. Report of Test Results

The initial Type A test shall be the subject of a summary technical report submitted to the Commission after the conduct of the test. This report shall include a schematic arrangement of the leakage rate measurement system, the instrumentation used, the supplemental test method, and the test program selected as applicable to the initial test, and all subsequent periodic tests. The report shall contain an analysis and interpretation of the leakage rate test data to the extent necessary to demonstrate the acceptability of the containment's leakage rate in meeting the acceptance criteria.

Periodic test leakage rate results of Type A, B, and C tests that meet the acceptance criteria shall be reported in the licensee's operating report. Leakage test results of Type A, B, and C tests that fail to meet the acceptance criteria shall be reported in a separate summary that includes an analysis and interpretation of the test data, the least-squares fit analysis of the test data, the instrumentation error analysis, and the structural conditions of the containment or components, if any, which contributed to the failure in meeting the acceptance criteria. Results and analyses of the supplemental verification test employed to demonstrate the validity of the leakage rate test measurements shall also be included.

(6) Recirculation Heat Removal Systems

a. Testing Requirements

The portion of the shutdown cooling system that is outside the containment shall be tested either by use in normal operation or hydrostatically tested at 250 psig at the interval specified in the 3.5(3)a(iii).

leakage rate and duplicates the pre-operational leak rate test of 30 psig. The specification provides relationships for relating in a conservative manner the measured leakage of air at 30 psig to the potential leakage of a steam-air mixture at 60 psig and 288°F. The specification also allows for possible deterioration of the leakage rate between tests, by requiring that only 70% or 80% of the allowable leakage rates actually be measured. The basis for the deterioration allowances is arbitrary judgments which are believed to be conservative and which will be confirmed or denied by periodic testing. If indicated to be necessary, the deterioration allowances will be altered based on experience. The durations for the integrated leakage rate tests are established to provide a minimum level of accuracy and to allow for daily cyclic variation in temperature and thermal radiation.

The frequency of the periodic integrated leakage rate test is keyed to the refueling outage schedule for the reactor, because these tests can best be performed during refueling shutdowns. The initial core loading is designed for approximately 12 months of power operation; thus, the first refueling outage will occur approximately 18 months after initial criticality. Subsequent refueling outages are scheduled at approximately 12-month intervals, although larger periods may be utilized.

The specified frequency of periodic integrated leakage rate tests is based on three major considerations. First is the low probability of leaks in the liner because of the test of the leak-tightness of the welds during erection and conformance of the complete containment to a low leak rate at 60 psig during pre-operational testing, which is consistent with 0.1% leakage at design basis accident conditions and absence of any significant stresses in the liner during reactor operation. Second is the more frequent testing, at the full accident pressure, of those portions of the containment envelope that are most likely to develop leaks during reactor operation (penetrations and isolation valves) and the low value (0.60) of the total leakage that is specified as acceptable from penetrations and isolation valves. Third is the tendon stress surveillance program, which provides assurance that an important part of the structural integrity of the containment is maintained.

The limiting leakage rates from the shutdown cooling system are judgment values based primarily on assuring that the components could operate without mechanical failure for a period on the order of 200 days after a design basis accident. The test pressure (250 psig) achieved either by normal system operation or by hydrostatic testing gives an adequate margin over the highest pressure within the system after a design basis accident. Similarly, the hydrostatic test pressure for the return lines from the containment to the shutdown cooling system (100 psig) gives an adequate margin over the highest pressure within the lines after a design basis accident.

A shutdown cooling system leakage of one gpm will limit off-site exposures due to leakage to insignificant levels relative to those calculated for direct leakage from the containment in the design basis accident. The safety injection system pump rooms are equipped with individual charcoal filters which are placed into operation by means of switches in the control room. The radiation detectors in the auxiliary building exhaust duct are used to detect high radiation level. The one gpm leak rate is sufficiently high to permit prompt detection and to allow for reasonable leakage through the pump seals and valve packings, and yet small enough to be readily handled by the pumps and radioactive waste system. Leakage to the safety injection system pump room sumps will be returned to the spent regenerant tanks. Additional makeup water to the containment sump inventory can be readily accommodated via the charging pumps from either the SIRW tank or the concentrated boric acid storage tanks.

In case of failure to meet the acceptance criteria for leakage from the shutdown cooling system or the penetrations, it may be possible to effect repairs within a short time. If so, it is considered unnecessary and unjustified to shutdown the reactor. The times allowed for repairs are consistent with the times developed for other engineered safeguards components.

A reduction in prestressing force and changes in physical conditions are expected for the prestressing system. Allowances have been made in the reactor building design for the reduction and changes. The inspection results for each tendon shall be recorded on the forms provided for that purpose and comparison will be made with the previous test results and the initial quality control records. Force-time trend lines will also be established and maintained for each of the surveillance tendons.

If the force-time trend line, as extrapolated, falls below the predicted force-time curve for one or more surveillance tendons, then before the next scheduled surveillance inspection, an investigation shall be made to determine whether the rate of force reduction is indeed occurring for other tendons. If the rate of reduction is confirmed, the investigation shall be extended so as to identify the cause of the rate of force reduction. The extension of the investigation shall determine the needed changes in the surveillance inspection schedule and the criteria and initial planning for corrective action. If the force-time trend lines of the surveillance tendons at any time exceed the upper bound curve of the band on the force-time graph, an investigation shall be made to determine the cause.

JUSTIFICATION FOR FEE CLASSIFICATION

The proposed amendment is deemed to be Class III, within the meaning of 10 CFR 170.22, in that it involves a single safety concern.

ATTACHMENT C