ATTACHMENT

Consumers Power Company
Palisades Plant
Docket 50-255

PROPOSED TECHNICAL SPECIFICATION PAGE CHANGES

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10 Pages

TSP0887-0139-NL04

4.5.3. Recirculation Heat Removal Systems

(3) Visual inspection shall be made for excessive leakage from components of the system at the interval specified in 6.15. Any significant leakage shall be measured by collection and weighing or by another equivalent method.

b. Acceptance Criterion

The maximum allowable leakage from the recirculation heat removal systems' components (which include valve stems, flanges and pump seals) shall not exceed 0.2 gallon per minute under the normal hydrostatic head from the SIRW tank (approximately 44 psig).

c. Corrective Action

Repairs shall be made as required to maintain leakage within the acceptance criterion of 4.5.3b.

4.5.4 Surveillance for Prestressing System

a. Tendon inspection shall be accomplished at five-year intervals for the life of the plant. The scheduled inspection dates for all subsequent inspections may be varied by not more than plus or minus one year from the base schedule.

b. The surveillance tendons shall be randomly but representatively selected from each of the following groups:

1. A minimum of 4 dome tendons including one from each dome tendon group.
2. A minimum of 4 vertical tendons.
3. A minimum of 5 hoop tendons.

For each inspection, the tendons shall be selected on a random basis except that those tendons whose routing has been modified to clear penetrations shall be excluded from the sample.

c. During each tendon inspection, the following field testing shall be performed:

1. Lift-off readings shall be taken for each of the surveillance tendons. The tests shall include the following actions:

(a) One tendon, randomly selected from each group of tendons during each inspection, shall be subjected to essentially complete detensioning to identify broken or damaged wires.

(b) The simultaneous measurement of elongation and jacking force during retensioning shall be made at a minimum of three approximately equally spaced levels of force between the seating force and zero.

2. While the tendon is in the detensioned state, each wire in the tendon will be checked for continuity.

3. Three wires, one from each of a vertical, a hoop and a dome tendon will be removed and identified for inspection. At each successive surveillance, the wires will be selected from different tendons. Each of the inspection wires removed will be visually inspected for evidence of corrosion or other deleterious effects and samples taken for laboratory testing.

4. The sheathing filler shall be inspected visually for color and coverage and samples shall be obtained for laboratory testing.

5. Tendon anchorage hardware such as bearing plates, stressing washers, shims and buttonheads shall be visually inspected for evidence of corrosion or other deleterious effects.

d. Following the field testing of 4.5.4c, the following laboratory testing shall be done:

1. Three tensile test specimens shall be cut from each of the three inspection wires removed (one from each end and one from the middle). One additional specimen shall be cut from the wire determined by field visual inspection to have the greatest amount of corrosion. Each of the wire samples shall be tested for ultimate strength, yield strength, and elongation.

2. The sheathing filler samples shall be taken from each end of each tendon examined. Vertical tendon samples shall be taken from the lower end. Samples shall be thoroughly mixed and analyzed for reserve alkalinity, water content, and concentration of water soluble chlorides, nitrates, and sulfides. Analyses shall be performed in accordance with the procedures and within the acceptance limits specified in ASME Code Section XI, Table IWL-2525-1.

4.5.4 Surveillance for Prestressing System (Contd)

Procedures shall be established to minimize voids and to assure that the volume of sheathing filler removed has been replaced upon completion of the inspection and amounts documented.

e. Acceptance criteria shall be as follows:

1. The average of all measured tendon forces for each type of tendon shall be equal to or greater than the minimum required prestress level, of 584 kips per tendon for dome tendons and, 615 kips per tendon for hoop and vertical tendons. The measured force in each individual tendon shall not be less than 95% of the predicted force, or

   (a) the measured force in not more than one tendon is between 90% and 95% of the predicted force, and

   (b) The measured forces in two tendons located adjacent to the tendon in (a) above are not less than 95% of the predicted forces, and

   (c) the measured forces in all the remaining sample tendons are not less than 95% of the predicted force.

   If measured force in any tendon is less than 90% of its predicted force, the tendon shall be completely detensioned and a determination shall be made as to the cause of such an occurrence and corrective action shall be taken. In addition, all such tendons shall have their forces measured as additional tendons in the next scheduled inspection period. The Commission shall be notified in accordance with Paragraph 4.5.4f.

2. Inspection wires shall indicate no significant loss of section by corrosion or pitting.

3. Tensile test specimens cut from inspection wires shall be tested for ultimate strength. Failure at less than 11.78 kips of any one of the test samples requires the Commission be notified in accordance with specification 4.5.4f.

4. Tendon anchorage hardware shall be free of significant corrosion, pitting, cracks or other deleterious effects.

4.5.4 Surveillance for Prestressing System (Contd)

f. If any element of the prestressing system fails to meet the acceptance criteria of 4.5.4e., the reporting provision of Specification 6.9.2 shall apply.

4.5.5 End Anchorage Concrete Surveillance

a. A VT-1 visual examination shall be performed on the end anchorage concrete surface at the surveillance tendon anchor points for signs of cracking, popouts, spalling, or corrosion. Concrete cracks having widths greater than 0.010 inches shall be evaluated and documented.

b. The end anchorage concrete surveillance inspection interval shall be the same as tendon surveillance interval.

c. Acceptance criteria

1. Crack widths shall be measured by using optical comparators or wire feeler gauge. Movements shall be measured by using demountable mechanical extensometers.

2. Concrete anchorage areas are acceptable if no concrete cracks are wider than 0.010 inches and no signs of new or progressive deterioration since the previous inspection are found.

3. Concrete surface conditions exceeding those stated in 4.5.5c.2 above shall be evaluated for the effect on tendon and containment structural integrity. The results of evaluation shall be included in the final surveillance report.

4.5.6 Liner Plate Surveillance - Deleted

4.5.7 Penetration Surveillance - Deleted

4.5.8 Dome Delamination Surveillance

If, as a result of a prestressing system inspection under Section 4.5.4, corrective retensioning of five percent (8) or more of the total number of dome tendons is necessary to restore their liftoff forces to within the limits of Specification 4.5.4, a dome delamination inspection shall be performed within 90 days following such corrective retensioning. The results of this inspection shall be reported to the NRC.

Basis

The containment is designed for an accident pressure of 55 psig.[1] While the reactor is operating, the internal environment of the containment will be air at approximately atmospheric pressure and a temperature of about 104°F. With these initial conditions, following a LOCA, the temperature of the steam-air mixture at the peak accident pressure of 55 psig is 283°F.

Prior to initial operation, the containment was strength-tested at 63 psig and then leak rate tested. The design objective of this preoperational leak rate test was established as 0.1% by weight per 24 hours at 55 psig. This leakage rate is consistent with the construction of the containment,[2] which is equipped with independent leak-testable penetrations and contains channels over all unaccessible containment liner welds, which were independently leak-tested during construction.

Accident analyses have been performed on the basis of a leakage rate of 0.1% by weight per 24 hours. With this leakage rate and with a reactor power level of 2530 MWt, the potential public exposure would be below 10 CFR 100 guideline values in the event of the Maximum Hypothetical Accident.[3]

The performance of a periodic integrated leak rate test during plant life provides a current assessment of potential leakage from the containment in case of an accident that would pressurize the interior of the containment. In order to provide a realistic appraisal of the integrity of the containment under accident conditions, this periodic leak rate test is to be performed without preliminary leak detection surveys or leak repairs and containment isolation valves are to be closed in the normal manner.

This normal manner is a coincident two-of-four high radiation or two-of-four high containment pressure signals which will close all containment isolation valves not required for engineered safety features except the component cooling lines' valves which are closed by SIS. The control system is designed on a two-channel (right and left) concept with redundancy and physical separation. Each channel is capable of initiating containment isolation.[4]

The test pressure of 28 psig for the periodic integrated leak rate test is sufficiently high to provide an accurate measurement of the leakage rate and it duplicates the preoperational leak rate test at 28 psig. The specification provides relationships for relating in a conservative manner the measured leakage of air at 28 psig to the potential leakage of a steam-air mixture at 55 psig and 283°F. The specification also allows for possible deterioration of the leakage rate between tests by requiring that only 75% of the allowable leakage rates actually be measured. The basis for these deterioration allowances is 10 CFR Part 50, Appendix J which is believed to be conservative and will be confirmed or denied by periodic testing. If indicated to be necessary, the deterioration allowances will be altered based on experience.

The duration of 24 hours for the integrated leak rate test is 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 leak rate test is keyed to the refueling schedule for the reactor because these tests can best be performed during refueling shutdowns. The specified frequency is as specified in 10 CFR Part 50, Appendix J which is based on three major considerations. First is the low probability of leaks in the liner because of (a) the test of the leak tightness of the welds during erection; (b) conformance of the complete containment to a low leak rate at 55 psig during preoperational testing which in consistent with 0.1% leakage at design basis accident (DBA) conditions: and (c) 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 ($0.60L_a$) 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 basis for specification of a total leakage rate of $0.60L_a$ from penetrations and isolation valves is specified to provide assurance that the integrated leak rate would remain within the specified limits during the intervals between integrated leak rate tests. This value allows for possible deterioration in the intervals between tests. 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 DBA. The test pressure (270 psig) achieved either by normal system operation or by hydrostatically testing gives an adequate margin over the highest pressure within the system after a DBA. 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 DBA.[5]

A shutdown cooling system leakage of 1/5 gpm will limit off-site exposures due to leakage to insignificant levels relative to those calculated for leakage directly from the containment in the DBA. The engineered safeguards room ventilation system is equipped with isolation valves which close upon a high radiation signal from a local radiation detector. These monitors shall be set at 2.2×10^5 cpm, which is well below the expected level, following a loss-of-coolant accident (LOCA), even without clad failure. The 1/5 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 sumps and radioactive waste system. Leakage to the engineered safeguards room sumps will be returned to the containment clean water receiver following a LOCA, via the equipment drain tank and pumps. 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.

4.5 CONTAINMENT TESTS (Cont'd)

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 shut down the reactor. The times allowed for repairs are consistent with the items developed for other engineered safety feature components.

A reduction in prestressing force and change 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 inspected shall be recorded on the forms provided for that purpose and comparison will be made with previous test results and the initial quality control records.

Force-time records will be established and maintained for each of the tendon groups, dome, hoop and vertical. If the force measured for a tendon is less than the lower bound curve of the force-time graph, two adjacent tendons will be tested. If either of the adjacent or more than one of the original sample population falls below the lower bound of the force-time graph, an investigation will be conducted before the next scheduled surveillance. The investigation shall be made to determine whether the rate of force reduction in 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 measured for a tendon at any time exceeds the upper bound curve of the band on the force-time graph, an investigation shall be made to determine the cause.

If the comparison of corrosion conditions, including chemical tests of the corrosion protection material, indicate a larger than expected change in the conditions from the time of installation or last surveillance inspection, and investigation shall be made to detect and correct the causes.[6]

4.5 CONTAINMENT TESTS (Cont'd)

The prestressing system is a necessary strength element of the plant safeguards and it is considered desirable to confirm that the allowances are not being exceeded. The technique chosen for surveillance is based upon the rate of change of force and physical conditions so that the surveillance can either confirm that the allowances are sufficient, or require maintenance before minimum levels of force or physical conditions are reached.

The end anchorage concrete is needed to maintain the prestressing forces. The design investigations concluded that the design is adequate. The prestressing sequence has shown that the end anchorage concrete can withstand loads in excess of those which result when the tendons are anchored. At the time of initial pressure testing, the containment building had been subjected to temperature gradients equivalent to those for normal operating conditions while the prestressing tendon loads are at their maximum.

However, after the initial pressure test both concrete creep and prestressing losses increase with the greatest rapidity and result in a redistribution of the stresses and a reduction in end anchor force. Because of the importance of the containment and the fact that the design was new, it was considered prudent to continue the surveillance after the initial period.(7)

4.5 CONTAINMENT TESTS (Cont'd)

Containment dome delamination inspections performed in 1970 and 1982 have confirmed that no concrete delamination has occurred. The possibility that delamination might occur in the future is remote because dome tendon prestress forces gradually diminish through normal tendon relaxation and concrete strength normally increases over time. To account for this remote possibility, however, an additional delamination inspection will be performed in the event that 5% or more of the installed tendons must be retensioned to compensate for excessive loss of prestress. This inspection would be to confirm that any systematic excessive prestress loss did not result from delamination and that the retensioning process did not result in delamination.

References

(1) FSAR, Section 5.1.2; Updated FSAR section 5.8.2.

(2) FSAR, Section 5.1.8; Updated FSAR section 5.8.8

(3) FSAR and Updated FSAR 14.22

(4) FSAR, Section 8.5.4; Updated FSAR Section 8.5.1.2

(5) FSAR and Updated FSAR Section 6.2.3

(6) FSAR, Section 5.1.8.4; FSAR, Amendment No 14, Question 5.37; and Updated FSAR Section 5.8.8.3.

(7) Updated FSAR, Section 5,8.8.6

(8) 10 CFR Part 50, Appendix J.