UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

OMAHA PUBLIC POWER DISTRICT

DOCKET NO. 50-285

FORT CALHOUN STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 97
License No. DPR-40

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Omaha Public Power District (the licensee) dated January 26, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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2. Accordingly, Facility Operating License No. DPR-40 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Facility Operating License No. DPR-40 is hereby amended to read as follows:

   B. Technical Specifications

      The Technical Specifications contained in Appendix A, as revised through Amendment No. 97 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective within 30 days of date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

A C Thadani

Ashok C. Thadani, Director
PWR Project Directorate #8
Division of PWR Licensing-B

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 3, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 97

FACILITY OPERATING LICENSE NO. DPR-40

DOCKET NO. 50-285

Revise Appendix "A" Technical Specifications as indicated below. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove Pages	Insert Pages
ii	ii
3-44	3-44
3-45	3-45
3-46	3-46
3-47	3-47
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3-51	3-51
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The report shall contain an analysis and interpretation of the Type A test results and a summary analysis of periodic Type B and Type C tests that were performed since the last Type A test.

Leakage test results from Type A, B, and C tests that failed to meet the applicable acceptance criteria shall be reported in a separate summary report approximately 3 months after the conduct of these tests. The Type A test report shall include an analysis and interpretation of the test data, the least-squares fit analysis of the test data (Type A tests only), the instrumentation error analysis (Type A tests only), 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.

3.0 SURVEILLANCE REQUIREMENTS

3.5 Containment Tests (Continued)

(7) Surveillance for Prestressing System

a. Surveillance Requirements

Two hundred ten dome tendons and 616 wall tendons shall be periodically inspected for symptoms of material deterioration or force reduction. Inspections will be performed on three dome tendons, one from each layer, and on three wall tendons of each orientation.

The surveillance tendons shall be inspected as follows:

(i) Lift-off readings shall be taken on each of the tendons selected to determine the load existing in the tendon at the time of inspection. At each surveillance period, readings may also be taken on the load cells of the special instrumented tendons. Force reductions on the surveillance tendons and on the instrumented tendons will be compared. If good correlation exists between these two groups of tendons through several surveillance periods, consideration will be given to eliminating some lift-off readings and monitoring of the load cells as an alternative. Each selected tendon shall be completely detensioned and examined for broken wires and any evidence of damage or deterioration of anchorage hardware.

(ii) One wire from each of three helical tendons and one wire of a dome tendon shall be removed. Each removed wire shall be carefully examined over its entire length for evidence of corrosion or other deleterious effects. Tensile tests shall be made on at least three samples cut from each of the four wires removed, one at each end and one at midlength, the samples being of a maximum length practical for testing. In special cases, the use of fatigue tests and accelerated corrosion tests may be considered.

(iii) Comparisons shall be made between the quality control records and each of the surveillance inspection records for each of the surveillance tendons.

After completion of the tendon surveillance the individual detensioned tendons shall be retensioned to a force commensurate with the average wire stress indicated by the last lift-off reading for that tendon.

3.0 SURVEILLANCE REQUIREMENTS

3.5 Containment Tests (Continued)

b. Acceptance Criteria

(i) The tendon force determined by the lift-off test shall be considered adequate if it is not less than the force shown on the appropriate lower limit curve of USAR Figure 5.10-4, as adjusted for wire removal, for the elapsed time between the original prestressing and the particular surveillance period. These lower limit curves have been generated by calculating the difference between the anticipated tendon force at end of plant life and the minimum tendon force to meet the design requirements. One half of this difference has been added to the anticipated total loss of prestress at the end of plant life and the curves have been drawn to meet this limit. Since the lock-off force on individual tendons is varied to compensate for elastic shortening of the structure, the tendon force at 70% of ultimate strength, rather than the actual lock-off force shall be taken as the initial prestress force. An allowable limit of not more than one defective tendon out of the total sample population is acceptable, provided an adjacent tendon on each side of the defective tendon is tested and is found to meet criteria. Should one of the adjacent tendons be also found defective, the Commission shall be notified in accordance with Regulatory Guide 1.16, "Reporting of Operating Information".

(ii) No unexpected change in corrosion conditions or grease properties.

(iii) All three tensile tests on any one wire indicate an ultimate strength at least equal to the specified minimum ultimate strength of the wire. If a single test on any one wire shows an ultimate strength less than the specified minimum, the Commission will be notified in accordance with Regulatory Guide 1.16, "Reporting of Operating Information".

c. Corrective Action

If the above acceptance criteria are not met, an immediate investigation shall be made to determine the cause(s) for the non-conformance to the criteria, and results will be reported to the Commission within 90 days.

d. Test Frequency

The tendons in the prestressing system shall be inspected once every 5 years.

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Basis

The containment is designed for an accident pressure of 60 psig.[2] While the reactor is operating, the internal environment of the containment will be air at approximately atmospheric pressure and a maximum temperature of about 120°F. With these initial conditions the temperature of the steam-air mixture at the peak accident pressure of 60 psig is 288°F.

Prior to initial operation, the containment was strength-tested at 69 psig and then was leak tested. The design objective of the pre-operational leakage rate test has been established as 0.1% by weight for 24 hours at 60 psig. This leakage rate is consistent with the construction of the containment, which is equipped with independent leak-testable penetrations and contains channels over all inaccessible containment liner welds, which were independently leak-tested during construction.

Safety analyses have been performed on the basis of a leakage rate of 0.1% of the free volume per day of the first 24 hours following the maximum hypothetical accident. With this leakage rate, a reactor power level of 1500 MWt, and with minimum containment engineered safety systems for iodine removal in operation (one air cooling and filtering unit), the public exposure would be well below 10 CFR Part 100 values in the event of the maximum hypothetical accident.[3] The performance of a periodic integrated leakage rate

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.

3.0 SURVEILLANCE REQUIREMENTS

3.5 Containment Tests (Continued)

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

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 have concluded that the design is adequate and this has been confirmed by tests. The prestressing sequence has shown that the end anchorage concrete can withstand loads in excess of those which result when the tendons are anchored. Further, the containment building was pressure tested to 1.15 times the maximum design pressure.

3.0 SURVEILLANCE REQUIREMENTS

3.16 Recirculation Heat Removal System Integrity Testing

Applicability

Applies to determination of the integrity of the shutdown cooling system and associated components.

Objective

To verify that the leakage from the recirculation heat removal system components is within acceptable limits.

Specifications

(1) a. The portion of the shutdown cooling system that is outside the containment shall be tested at 250 psig at each refueling outage, or other convenient intervals, but in no case at intervals greater than 2 years.

b. Piping from valves HCV-383-3 and HCV-383-4 to the discharge isolation valves of the safety injection pumps and containment spray pumps shall be hydrostatically tested at no less than 100 psig at the testing frequency specified in (1)a. above.

c. Visual inspection of the system's components shall be performed at the frequency specified in (1)a. above to uncover any significant leakage. The leakage shall be measured by collection and weighing or by any other equivalent method.

(2) a. The maximum allowable leakage from the recirculation heat removal system's components (which include valve stems, flanges, and pump seals) shall not exceed one gallon per minute, under the normal hydrostatic head from the SIRW tank.

b. Repairs shall be made as required to maintain leakage within the acceptable limits.

Basis

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.[1] 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.

3.0 SURVEILLANCE REQUIREMENTS

3.16 Recirculation Heat Removal System Integrity Testing (Continued)

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.[2] 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 associated components, 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.

References

(1) USAR, Section 9.3

(2) USAR, Section 6.2