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. 95
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 March 30, 1977, as supplemented by letters dated May 15, 1980, September 3 and November 5, 1982, January 26, 1983 and November 27, 1985 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.

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. 95 , 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 the 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: February 3, 1986

ATTACHMENT TO LICENSE AMENDMENT NO. 95

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
3-37	3-37
3-38	3-38
3-39	3-39
3-40	3-40
3-41	3-41
3-42	3-42
3-43	3-43
3-44	3-44
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3-49	3-49
3-50	3-50
3-53	3-53
Reference 3	3-53a

3.0 SURVEILLANCE REQUIREMENTS

3.5 Containment Test

Applicability

Applies to containment leakage and structural integrity.

Objective

To verify that the:

(1) potential leakage from containment is within acceptable limits, and

(2) structural performance of all important components in the containment prestressing system is acceptable.

Specifications

(1) Containment Building Leak Rate Tests

Tests shall be conducted to assure that leakage of the primary reactor containment and associated systems is maintained within allowable leakage rate limits. Periodic surveillance shall be performed to assure proper maintenance and leak repair of the containment structure and penetrations during the plant's operating life.

Definitions of terms used in the leak rate testing specifications:

Leakage rate - for test purposes is that leakage of containment air which occurs in a unit of time. Stated as a percentage of weight of the original content of containment air at the leakage rate test pressure that escapes to the outside atmosphere during a 24 hour test period.

Maximum allowable leakage rate (L_a) - the design basis leakage rate of 0.1% by weight of the containment atmosphere per 24 hours at a pressure of 60 psig.

Overall integrated leakage rate - that leakage rate which is obtained from a summation of leakage through all potential leakage paths including containment welds, valves, fittings, and components which penetrate containment.

Acceptable criteria - the standard against which test results are to be compared for establishing the functional acceptability of the containment as a leakage limiting boundary.

(2) Integrated Leak Rate Test (Type A Test)

a. Introduction

Type A tests are intended to measure the reactor containment overall integrated leakage rate at periodic intervals.

b. Pretest Requirements

A general inspection of the accessible interior and exterior surfaces of the containment structures and components shall be performed prior to any Type A test to uncover any evidence of structural deterioration which may affect either the containment structural integrity or leak-tightness. If there is evidence of structural deterioration, the Type A tests shall not be performed until corrective action is taken in accordance with repair procedures, non-destructive examinations, and tests as specified in the applicable code specified in 10 CFR Part 50.55a at the commencement of repair work. Such structural deterioration and corrective actions taken shall be reported as part of the Type A test report.

During the period between the initiation of the containment inspection and performance of the Type A test, no repairs or adjustments shall be made so that the containment can be tested in as close to the "as is" condition as practical. During the period between the completion of one Type A test and the initiation of the containment inspection for the subsequent Type A test, repairs or adjustments shall be made to components whose leakage exceeds that specified in the Technical Specifications as soon as practical after identification. This requirement is interpreted not to preclude performance of Type B and Type C testing and required repairs prior to initiation of the containment inspection and the performance of the Type A test.

If during a Type A test, potentially excessive leakage paths are identified which interfere with satisfactory completion of the test, or which result in the Type A test not meeting the acceptance criteria, the Type A test shall be temporarily suspended. Thereafter, repairs and/or adjustments to equipment shall be made and the Type A test resumed. The corrective action taken, the change in leakage rate resulting from the repairs and overall integrated leakage determined from the Type A and local leak rate tests shall be included in a report submitted to the Commission.

Closure of containment isolation valves for the Type A test shall be accomplished by normal operation and without any preliminary exercising or adjustments (e.g., no tightening of valve after closure by valve motor). Repairs of maloperating or leaking valves shall be made as necessary. Information on any valve closure malfunction or valve leakage that requires corrective action before the test, shall be included in the Type A Leak Test Report submitted to the Commission.

The containment test conditions shall stabilize for a period of approximately 4 hours prior to the start of the leakage rate test.

Those portions of the fluid systems that are part of the reactor coolant pressure boundary and are open directly to the containment atmosphere under post-accident conditions and become an extension of the boundary of the containment shall be opened or vented to the containment atmosphere prior to and during the test. Portions of closed systems inside containment that penetrate containment and rupture as a result of a loss of coolant accident shall be vented to the containment atmosphere. All vented systems shall be drained of water or other fluids to the extent necessary to assure exposure of the system containment isolation valves to containment air test pressure and to assure they will be subjected to the post-accident differential pressure. Systems that are required to maintain the plant in a safe condition during the test shall be operable in their normal mode, and need not be vented. Systems that are normally filled with water and operating under post-accident conditions, such as the containment heat removal system and the component cooling water system, need not be vented. However, the containment isolation valves in the systems defined in this section shall be tested in accordance with Section 3.5(4). The measured leakage rate from these tests shall be reported to the Commission.

c. Test Methods

All Type A tests shall be conducted in accordance with the provisions of 10 CFR Part 50, Appendix J.

The accuracy of any Test A shall be verified by a supplemental test. The supplemental test method selected shall be conducted for sufficient duration to establish accurately the change in leakage rate between the Type A test and the supplemental Type A test. Results from the supplemental test are acceptable provided the difference between the supplemental test data and the Type A test data is within 0.25 L_a. If results are not within 0.25 L_a, the reason shall be determined, corrective action taken, and a successful supplemental test performed.

Test leakage rates shall be calculated using absolute values corrected for instrument error.

d. Acceptance Criteria

The maximum allowable leakage rate shall not exceed 0.1%.

The total measured leakage rate at a pressure of 60 psig shall be less than 0.75 L_a. If local leakage measurements are taken to effect repairs in order to meet 0.75 L_a acceptance criteria, these measurements shall be taken at a pressure of 60 psig.

If two consecutive Type A tests fail to meet the acceptance criteria, notwithstanding the requirements of the testing frequency, a Type A test shall be performed at each refueling outage or approximately every 18 months, whichever occurs first, until two consecutive Type A tests meet the acceptance criteria, after which time the normal testing frequency schedule may be resumed.

e. Testing Frequency

A set of three Type A tests shall be performed, at approximately equal intervals during each 10 year service period. The third test of each set shall be conducted when the plant is shutdown for the 10-year in-service inspections.

The performance of Type A tests shall be limited to periods when the plant facility is non-operational and secured in the shutdown condition under administrative control and in accordance with the safety procedures defined in the license.

(3) Containment Penetrations Leak Rate Tests (Type B Tests)

a. Introduction

Type B tests are intended to detect local leaks and to measure leakage across each pressure-containing or leakage limiting boundary for the containment penetrations.

b. Test Methods

Type B tests shall be performed by local pneumatic pressurization of the containment penetrations, either individually or in groups, at a pressure of 60 psig.

Examination shall be performed by halide leak-detection method or by other equivalent test methods such as measurement of the rate of makeup required to maintain the test volume at 60 psig.

c. Acceptance Criteria

The combined leakage rate of all penetrations and valves subject to Type B and Type C tests shall be less than or equal to 0.6 L_a.

If at any time it is determined that a leakage rate is greater than 0.6 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.

The results of personnel access lock door seal tests at 5 psig shall not exceed .01 L_a.

d. Testing Frequency

Type B tests shall be performed during each refueling outage, or other convenient intervals, but in no case at intervals greater than 2 years, except the personnel access lock (PAL) which will be tested as follows:

(i) Every six months the entire PAL assembly shall be leak tested at 60 psig.

(ii) If the PAL is opened during periods when containment integrity is not required, the PAL door seals shall be leak tested at 5 psig at the end of such periods and the entire PAL assembly shall then be leak tested at 60 psig within two weeks of achieving the required condition for containment integrity.

(iii) If the PAL is opened during the interval between the six-month tests when containment integrity is required, the PAL door seals shall be leak tested at a pressure not less than 5 psig within 72 hours. If the PAL is opened more frequently than once per 72 hours, the door seals shall be leak tested at a pressure of 5 psig at least once every 72 hours during the period of frequent openings.

e. Penetrations to be Tested (1)

(i) Equipment Hatch

(ii) Personnel Access Lock

(iii) Mechanical Penetrations M-1 through M-99

(iv) Fuel Transfer Tube (Mechanical Penetration M-100)

(v) Electrical Penetrations

A-1	B-9	D-6	F-2	E-HCV-383-3A
A-2	B-10	D-7	F-4	E-HCV-383-3B
A-4	B-11	D-8	F-5	E-HCV-383-4A
A-5	C-1	D-9	F-6	E-HCV-383-4B
A-6	C-2	D-10	F-7	
A-7	C-4	D-11	F-8	
A-8	C-5	E-1	F-9	
A-9	C-6	E-2	F-10	
A-10	C-7	E-4	F-11	
A-11	C-8	E-5	G-1	
B-1	C-9	E-6	G-2	
B-2	C-10	E-7	G-3	
B-4	C-11	E-8	G-4	
B-5	D-1	E-9	H-1	
B-6	D-2	E-10	H-2	
B-7	D-4	E-11	H-3	
B-8	D-5	F-1	H-4	

(4) Containment Isolation Valves Leak Rate Tests (Type C Tests)

a. Introduction

Type C tests are intended to measure containment isolation valve leakage rates.

b. Test Methods

Type C tests shall be performed by local pressurization with air or nitrogen at a pressure of 60 psig. The pressure shall be applied in the same direction as that when the valve would be required to perform its safety function, unless it can be determined that the results from the tests for a pressure applied in a different direction will provide equivalent or more conservative results. 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).

c. Acceptance Criteria

The combined leakage rate of all penetrations and valves subject to Type B and Type C tests shall be less than or equal to 0.6 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 criterion.

If at any time it is determined that a leakage rate is greater than 0.6 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.

d. Testing Frequency

Type C tests shall be performed during each refueling outage, or other convenient intervals, but in no case at intervals greater than 2 years. The containment purge isolation valves shall also be leakage tested prior to bringing the reactor out of each cold or refueling shutdown but in no case at intervals greater than nine months. If the purge valves are opened during cold or refueling shutdown, the leak test shall be performed after the purge valves are closed for the last time.

e. Penetrations to be Tested(1)

M-2	M-31	M-52
M-7	M-38	M-53
M-8	M-39	M-57
M-11	M-40	M-58
M-14	M-42	M-69
M-15	M-43	M-73
M-18	M-44	M-74
M-19	M-45	M-79
M-20	M-46	M-80
M-22	M-47	M-87
M-24	M-48	M-88
M-25	M-50	M-HCV-383-3
M-30	M-51	M-HCV-383-4

(5) Special Testing Requirements

Any major modification or replacement of a component which is part of the containment boundary shall be followed by either Type A, Type B, or Type C tests as applicable for the area affected by the modification and shall meet the applicable acceptance criteria. Minor modifications, or replacements, performed directly prior to the conduct of a scheduled Type A test do not require a separate test.

(6) Report on Test Results

Leak rate tests shall be the subject of a summary technical report submitted to the Commission approximately 3 months after the conduct of each test. The report shall be titled "Reactor Containment Building Integrated Leak Rate Test."

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.

(7) 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).

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 interval specified in 3.5(3)a(iii).

Visual inspection shall be made for excessive leakage from components of the system. 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 one gallon per minute, under the normal hydrostatic head from the SIRW tank.

c. Corrective Action

Repairs shall be made as required to maintain leakage within the acceptance criterion of 3.5(3)a(ii).

d. Test Frequency

Tests of the recirculation heat removal system shall be conducted at each major refueling.

(8) Surveillance for Prestressing System

a. Surveillance Requirements

Two hundred ten dome tendons and 616 wall tendons are provided. These shall be periodically inspected for symptoms of material deterioration or force reduction. During each of the first three inspection periods, six dome tendons uniformly spaced in the three layers and seven helical wall tendons in each orientation spaced uniformly apart will be examined. Unless experience indicates otherwise, successive 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.

b. Acceptance Criteria

Acceptance criteria for the tendons are as follows:

(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 FSAR 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

Inspections shall be as follows:

(i) One year (from date of initial containment leak rate test).
(ii) Two years
(iii) Four years
(iv) Every five years thereafter for the life of the plant.

(9) End Anchorage Concrete Surveillance

a. Surveillance Requirements

Specific locations for surveillance will be chosen from the combined information from the design calculations; the as-built end concrete and prestressing records; observations of the end anchorage concrete during and after prestressing; and the results of strain and deformation measurements made during prestressing and the initial structural test.

The inspections made shall include:

(i) Visual inspection of the end anchorage concrete exterior surfaces.

(ii) A determination of the temperatures of the liner plate area or containment interior surface in locations nearest to the end anchorage concrete under surveillance.

(iii) Measurement of concrete temperatures at specific end anchorage concrete surfaces being inspected.

(iv) The mapping of the predominant visible concrete crack patterns.

(v) The measurement of the crack widths by use of optical comparators or wire feeler gauges.

(vi) The measurement of movements, if any, by use of demountable mechanical extensometers.

The measurements and observations shall be compared with those to which prestressed structures have been subjected in normal and abnormal load conditions and with those of preceding measurements and observations at the same location on the reactor building.

b. Acceptance Criterion

If the inspections determine that the conditions are favorable in comparison with experience and predictions, the close inspections will be terminated by the last of the inspections stated in the schedule and a report will be prepared which documents the findings and recommends the schedule for future inspections, if any. If the inspections detect symptoms of greater than normal cracking or movements, an immediate investigation will be made to determine the cause.

c. Test Frequency

The inspection intervals will be approximately one-half year and one year after the initial structural test and shall be chosen such that the inspection occurs during the warmest and coldest part of the year following the initial structural test.

(10) Liner Plate Surveillance

a. Surveillance Requirements

The liner plate will be visually examined before the initial pressure test to determine the following:

(i) Locate areas which have inward deformations. Measure and record inward deformations. The areas will be permanently marked for future reference.

(ii) Try to locate areas having strain concentrations by visual examination paying particular attention to the condition of the liner surface. Record the location of any areas having strain concentrations.

b. Acceptance Criterion

Shortly after the initial pressure test and at one year after initial start-up, reexamine the areas located in section (1). Measure and record inward deformations. Record observations pertaining to strain concentrations. If the difference in the measured inward deformations exceeds 0.25 inch (for a particular location) and/or changes in strain concentration exists, then an investigation will be made. The investigation will determine the cause and any necessary corrective action.

c. Test Frequency

The surveillance program will only be continued beyond the one year after initial start-up inspection if some corrective action is needed.

(11) Penetrations Surveillance

a. Surveillance Requirements

The penetration assemblies will be visually inspected before the initial pressure test, shortly after the initial pressure test and one year after initial start-up.

The inspection will place particular emphasis on the nozzle-to-penetration reinforcing plate weld and all other closure welds.

The inspection will concentrate on finding any indications which might affect the leak tightness or structural integrity such as weld cracks, pinholes, flaws, etc.

b. Acceptance Criteria

If any indications are found, further examination using non-destructive testing techniques will be used to verify or disprove the visual examination. All indications will be documented stating type of indication, location, NDT used for verification and final results.

c. Corrective Action

For any verified indication, an examination will be conducted to determine the cause and any necessary corrective action.

d. Frequency

The surveillance program will only be continued beyond the one year after initial start-up inspection if some corrective action is needed. The frequency of inspection for a continued surveillance program will be determined shortly after the one year after initial start-up inspection.

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

test during plant life provides a current assessment of potential leakage from the containment.

The reduced pressure (5 psig) test on the PAL is a conservative method of testing and provides adequate indication of any potential containment leakage path. The test is conducted by pressurizing between two resilient seals on each door. The test pressure tends to unseat the resilient seals which is opposite to the accident pressure that tends to seat the resilient seals. The six month test ensures the overall PAL integrity at 60 psig.

The frequency of the periodic integrated leakage rate test (Type A test) is keyed to the refueling schedule for the reactor, because this test can only be performed during refueling shutdowns.

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.

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.

References

(1) USAR, Section 5.9.1

(2) USAR, Section 5.1.1

(3) USAR, Section 14.15

Reference 3 entitled "Omaha Public Power District, Fort Calhoun Station Unit No. 1, Reactor Containment Building Leak Rate Test, Supplementary Report, June 1973" is deleted from the Technical Specifications with this amendment. This deletion encompasses the cover sheet, Table of Contents, and the six Reference 3 pages.