August 15, 2003

Mr. R. T. Ridenoure
Division Manager - Nuclear Operations
Omaha Public Power District
Fort Calhoun Station, FC-2-4 Adm.
P.O. Box 550
Fort Calhoun, NE 68023-0550

SUBJECT: FORT CALHOUN STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT (TAC NO. MB6473)

Dear Mr. Ridenoure:

The Commission has issued the enclosed Amendment No. 220 to Facility Operating License No. DPR-40 for the Fort Calhoun Station, Unit No. 1. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated October 8, 2002, as supplemented by letters dated April 11 and May 21, 2003.

The amendment grants a one-time five-year extension to the current ten-year test interval for the containment integrated leak rate testing.

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next biweekly *Federal Register* notice.

Sincerely,

/RA/

Alan B. Wang, Project Manager, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosures: 1. Amendment No. 220 to DPR-40
2. Safety Evaluation

cc w/encls: See next page

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Ft. Calhoun Station, Unit 1

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OMAHA PUBLIC POWER DISTRICT

DOCKET NO. 50-285

FORT CALHOUN STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 220
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 October 8, 2002, as supplemented by letters dated April 11 and May 21, 2003, 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 license 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. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 220, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance and shall be implemented within 60 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by Jack Donohew for/

Stephen Dembek, Chief, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: August 15, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 220

FACILITY OPERATING LICENSE NO. DPR-40

DOCKET NO. 50-285

Replace the following page of the Appendix A Technical Specifications with the attached revised page. The revised page is identified by amendment number and contains vertical lines indicating the areas of change.

REMOVE	INSERT
5-16	5-16

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 220 TO FACILITY OPERATING LICENSE NO. DPR-40

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION, UNIT NO. 1

DOCKET NO. 50-285

1.0 INTRODUCTION

By application dated October 8, 2002, as supplemented by letters dated April 11, 2003, and May 21, 2003, Omaha Public Power District (OPPD) requested changes to the Technical Specifications (Appendix A to Facility Operating License No. DPR-40) for the Fort Calhoun Station, Unit No. 1 (FCS). The requested change would grant a one-time five-year extension to the current ten-year test interval for the containment integrated leak rate testing (ILRT).

The supplemental letters dated April 11 and May 21, 2003, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on November 12, 2002 (67 FR 68742).

2.0 REGULATORY EVALUATION

Title 10 of the Code of Federal Regulations (CFR), Part 50, Appendix J, was revised in 1995 by the addition of Option B, "Performance-Based Requirements," to the original requirements, which were then designated as Option A, "Prescriptive Requirements." Option B requires that a Type A test be conducted at a periodic interval based on historical performance of the overall containment system. FCS Technical Specification (TS) 5.19 requires that leakage rate testing be performed as required by 10 CFR Part 50, Appendix J, Option B, as modified by approved exemptions, and in accordance with the guidelines contained in Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak-Test Program," with certain exceptions which are listed in the TS. This RG endorses, with certain exceptions, Nuclear Energy Institute (NEI) Report 94-01, Revision 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," dated July 26, 1995.

A Type A test is an overall (integrated) leakage rate test of the containment structure. NEI 94-01 specifies an initial test interval of 48 months, but allows an extended interval of 10 years, based upon two consecutive successful tests. There is also a provision for extending the test interval an additional 15 months in certain circumstances. The most recent two Type A tests at Fort Calhoun have been successful, so the current interval requirement is ten years.

The licensee is requesting an addition to TS 5.19, "Containment Leakage Rate Testing Program," which would add another exception from the guidelines of RG 1.163 regarding the Type A test interval. Specifically, the proposed TS states that the first Type A test performed after the November 1993 Type A test shall be no later than November 2008. The local leakage rate tests (Type B and Type C tests), including their schedules, are not affected by this request.

3.0 EVALUATION

The staff has reviewed the licensee's regulatory and technical analysis in support of its license amendment, which is described in the licensee's submittal dated October 8, 2002 and request for additional information responses dated April 11 and May 21, 2003. The following evaluation addresses the acceptability of this amendment request.

3.1 Applicable Technical Specification Requirements

For FCS, the licensee is proposing to revise the requirements in Section 5.19 of the Administrative Controls section of the TS for containment leakage by inserting item (4) in Section 5.19 as follows:

> 4. The first Type A test performed after the November 1993 Type A test shall be no later than November 2008.

3.2 Inservice Inspection (ISI) for Primary Containment Integrity

FCS utilizes a Combustion Engineering pressurized water reactor enclosed in a steel-lined prestressed concrete containment. The containment vessel consists of a continuous and essentially leak-tight steel membrane which includes the cylindrical portion, the torispherical portion, and the floor liner plate on the top of the basemat. The containment vessel is penetrated by access penetrations, process piping and electrical penetrations. The integrity of the penetrations and isolation valves are verified through Type B and Type C local leak rate tests (LLRTs) as required by 10 CFR Part 50, Appendix J. The overall leak-tight integrity of the primary containment is verified through ILRTs. These tests are performed to verify the essentially leak-tight characteristics of the containment at the design basis accident (DBA) pressure. The last ILRT was performed in November 1993. The next ILRT is scheduled during the outage in calendar year 2003. With the extension of the ILRT interval, the licensee will perform the next overall verification no later than November 2008. The amendment request did not provide any information on the condition of containment components. However, in response to the staff's request for additional information, the licensee provided information related to the inservice inspection of the containment and discussed potential areas of degradation in the containment that might not be apparent in the risk assessment. The staff's evaluation of the licensee's responses to the ISI-related issues is discussed below.

The licensee has implemented the containment inservice inspection requirements mandated by 10 CFR 50.55a using the requirements of the 1992 Edition and the 1992 Addenda of the American Society of Mechanical Engineers (ASME) Code, Section XI, Subsections IWE and IWL. Based on the summary of the examination procedures provided in the licensee's April 11, 2003, letter, the staff finds the licensee's program for examining the accessible

portions of the containment concrete and steel surfaces and post-tensioning system adequate for detecting flaws and degradation.

In its April 11, 2003, response to the staff's RAI on examination and testing of seals, gaskets, and pressure retaining bolts, the licensee stated the following:

- For penetrations, in general, FCS has not extended Type B testing beyond a five-year interval.

- In accordance with the FCS containment leakage rate testing program, access openings (fuel transfer flange, personnel access lock, and equipment hatch) are scheduled for Type B testing at every refueling outage after closure following outage operation.

The staff finds that the schedule for testing penetration seals, gaskets, and pressure retaining bolting is frequent enough to ensure that any degradation of these pressure retaining components will be detected in a timely manner, and, therefore, acceptable for the period of extended ILRT interval.

In the past, the staff has found that two-ply stainless steel bellows are susceptible to trans-granular stress corrosion cracking and leakage through them is not detectable by Type B testing (See NRC Information Notice 92-20, "Inadequate Leak Rate Testing"). In response to the staff's question on the ability of Type B testing to detect leakage from the pressure boundary bellows, OPPD provided the following response:

"Bellows forming containment boundaries at FCS are limited to those at the piping penetrations for Main Steam and Feedwater Systems. Type B testing is currently performed on all bellows at a Refueling Outage interval. Type B test results have been effective in identifying leakage and have been critical to determinations for bellows replacement. These bellows have significant design differences from those described in NRC Information Notice 92-20 in that they are not manufactured in layers (not tested between layers) which may inhibit effective Type B testing."

The licensee stated that the Type B testing performed on penetrations with bellows has been effective in identifying leakage and detecting degradation of bellows, and the frequency of the testing will enable the licensee to identify any future degradation of bellows. The staff finds that the implementation of the licensee's containment ISI program, including the areas subjected to subsequent inspections and testing, provides reasonable assurance that the identified degradation occurring in the accessible areas of the containments will be adequately monitored during the ILRT interval increase.

In response to the staff's question on whether the licensee is planning a major modification and repair of containment for facilitating steam generator or reactor vessel head replacement during the proposed extension period, the licensee provided the following response:

"OPPD is preparing to replace steam generators and possibly other reactor coolant system components within the next four years. These replacements will require constructing and repairing a containment wall opening larger than the existing equipment hatch. FCS

understands ASME Code Section XI requires an ILRT (pressure testing) after major repair/replacement activity to the containment structure. This relief request does not address any relief from the repair and replacement test requirements; it only requests relief from the periodic ILRT requirement. If FCS determines that alternative methods to ensure safe and reliable containment conditions are appropriate, separate relief from the containment structure repair/replacement testing requirement will be requested."

For the purpose of approving the proposed amendment request, the staff considers OPPD's explanation acceptable because this ILRT extension request, when approved, will not eliminate the need to perform an ILRT as required by other regulations (i.e., 10 CFR 50.55a).

In its May 21, 2003, response to the staff's question on incorporating the potential degradation in uninspectable areas of the containment in the risk assessment, OPPD considered the following steps in its risk assessment:

- The likelihood of a corrosion-related flaw in steel shell/liner was determined.

- The likelihood of a corrosion-related steel shell/liner flaw was adjusted for age.

- The change in flaw likelihood for an increase in leak rate testing interval was determined.

- The likelihood of a breach in containment for a given liner/shell flaw was determined.

- The likelihood of failure to detect a flaw by visual inspection was determined considering the portion of the liner/shell that is uninspectable.

The acceptance of OPPD's risk-assessment parameters and evaluation is discussed in Section 3.3 of this safety evaluation.

Based on its review of the information provided in the amendment request and responses to the staff's RAI, the staff finds that (1) the structural degradation of the accessible areas of the FCS containment will be adequately monitored through the periodic inservice inspection conducted as required by Subsection IWE of Section XI of the ASME Code, and (2) the integrity of the penetrations and containment isolation valves will be periodically verified through Type B and Type C tests as required by 10 CFR Part 50, Appendix J. In addition, the system pressure tests for containment pressure boundary (i.e., Appendix J tests, as applicable) are required to be performed following repair and replacement activities in accordance with Subarticle IWE-5000 and IWL-5000 of Section XI of the ASME Code. Significant degradation of the primary containment pressure boundary is required to be reported under 10 CFR 50.72 or 10 CFR 50.73.

3.3 Risk Impact Assessment

The licensee has performed a risk impact assessment of extending the Type A test interval to 15 years. The risk assessment was provided in the October 8, 2002, application for license amendment. Additional analysis and information was provided by the licensee in its letters

dated April 11 and May 21, 2003. In performing the risk assessment, the licensee considered the guidelines of NEI 94-01, the methodology used in Electric Power Research Institute (EPRI) TR-104285, "Risk Impact Assessment of Revised Containment Leak Rate Testing," and RG 1.174, "An Approach For Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."

The basis for the current ten-year test interval is provided in Section 11.0 of NEI 94-01, Revision 0, and was established in 1995 during the development of the performance-based Option B to Appendix J. Section 11.0 of NEI 94-01 states that NUREG-1493, "Performance-Based Containment Leak-Test Program," provided the technical basis to revise leakage rate testing requirements contained in Option B to Appendix J. The basis consisted of qualitative and quantitative assessments of the risk impact (in terms of increased public dose) associated with a range of extended leakage rate test intervals. To supplement this basis, industry undertook a similar study. The results of that study are documented in EPRI Research Project Report TR-104285.

The EPRI study used an analytical approach similar to that presented in NUREG-1493 for evaluating the incremental risk associated with increasing the interval for Type A tests. The Appendix J, Option A, requirements that were in effect for Fort Calhoun early in the plant's life, required a Type A test frequency of three tests in ten years. The EPRI study estimated that relaxing the test frequency from three tests in ten years to one test in ten years would increase the average time that a leak that was detectable only by a Type A test goes undetected from 18 to 60 months. Since Type A tests only detect about three percent of the leaks (the rest are identified during local leak rate tests based on industry leakage rate data gathered from 1987 to 1993), this results in a ten percent increase in the overall probability of leakage. The risk contribution of pre-existing leakage for the pressurized water reactor and boiling water reactor representative plants in the EPRI study confirmed the NUREG-1493 conclusion that a reduction in the frequency of Type A tests from 3 tests in 10 years to 1 test in 20 years leads to an "imperceptible" increase in risk that is on the order of 0.2 percent and a fraction of 1 person-rem per year in increased public dose.

Building upon the methodology of the EPRI study, the licensee assessed the change in the predicted person-rem/year frequency. The licensee quantified the risk from sequences that have the potential to result in large releases if a pre-existing leak were present. Since the Option B rulemaking in 1995, the staff has issued RG 1.174 on the use of probabilistic risk assessment (PRA) in evaluating risk-informed changes to a plant's licensing basis. The licensee has proposed using RG 1.174 guidance to assess the acceptability of extending the Type A test interval beyond that established during the Option B rulemaking.

RG 1.174 defines very small changes in the risk-acceptance guidelines as increases in core damage frequency (CDF) less than 10^{-6} per year and increases in large early release frequency (LERF) less than 10^{-7} per year. Since the Type A test does not impact CDF, the relevant criterion is the change in LERF. The licensee has estimated the change in LERF for the proposed change and the cumulative change from the original frequency of three tests in a ten year interval. RG 1.174 also discusses defense-in-depth and encourages the use of risk analysis techniques to help ensure and show that key principles, such as the defense-in-depth philosophy, are met. The licensee estimated the change in the conditional containment failure

probability for the proposed change to demonstrate that the defense-in-depth philosophy is met. The licensee provided analyses, as discussed below. The following comparisons of risk from a change in test frequency from 3 test in 10 years to 1 test in 15 years are considered to be bounding for the Fort Calhoun comparative frequencies of 1 test in 10 years to 1 test in 15 years. The following conclusions can be drawn from the analysis associated with extending the Type A test frequency:

1. Given the change from a 3-in-10 year test frequency to a 1-in-15 year test frequency, the increase in the total integrated plant risk, in person-rem per year, is estimated to be about 0.16 percent. This increase is comparable to that estimated in NUREG-1493, where it was concluded that a reduction in the frequency of tests from 3-in-10 years to 1-in-20 years leads to an "imperceptible" increase in risk. Therefore, the increase in the total integrated plant risk for the proposed change is considered small and supportive of the proposed change.

2. The increase in LERF resulting from a change in the Type A test frequency from the original 3-in-10 years to 1-in-15 years is estimated to be 4.2×10^{-8} per year based on the current plant risk model, which is an internal events PRA that also includes the major risk contributors from seismic events. There is some likelihood that the flaws in the containment estimated as part of the Class 3b frequency would be detected as part of the IWE/IWL visual examination of the containment surfaces (as identified in American Society of Mechanical Engineers [ASME] Boiler and Pressure Vessel Code, Section XI, Subsections IWE/IWL). The most recent visual examination of the Fort Calhoun containment was performed in 2001. The next scheduled IWE/IWL containment inspection is in 2003. Visual inspections are expected to be effective in detecting large flaws in the visible regions of containment, and this would reduce the impact of the extended test interval on LERF. The licensee's risk analysis considered the potential impact of age-related corrosion/degradation in inaccessible areas of the containment shell on the proposed change. The increase in LERF associated with corrosion events is estimated to be less than 1×10^{-8} per year. The staff concludes that increasing the Type A interval to 15 years results in only a small change in LERF and is consistent with the acceptance guidelines of RG 1.174.

3. RG 1.174 also encourages the use of risk analysis techniques to help ensure and show that the proposed change is consistent with the defense-in-depth philosophy. Consistency with the defense-in-depth philosophy is maintained if a reasonable balance is preserved between prevention of core damage, prevention of containment failure, and consequence mitigation. The licensee estimates the change in the conditional containment failure probability to be an increase of less than 0.1 percentage point for the cumulative change of going from a test frequency of 3-in-10 years to 1-in-15 years. The staff finds that the defense-in-depth philosophy is maintained based on the small magnitude of the change in the conditional containment failure probability for the proposed amendment.

Based on these conclusions, the staff finds that the increase in predicted risk due to the proposed change is within the acceptance guidelines while maintaining the defense-in-depth philosophy of RG 1.174 and, therefore, is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Nebraska State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or changes an inspection or a surveillance requirement . The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (67 FR 68742). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: J. Pulsipher
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Date: August 15, 2003