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. 57
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 May 19, 1980, 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 hereby amended as follows:

(1) Amend paragraph 3.B. to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 57, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(2) Delete paragraphs 3.C., 3.E., 3.F., and 3.G.

(3) Renumber existing paragraph 3.D. to read 3.C.

(4) Delete paragraphs 4.A., 4.B., 4.C., 4.D., 4.E., and 4.F.

(5) Renumber existing paragraph 5 to read 4.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachments:
1) Revised page 4 to DPR-40
2) Changes to the Technical Specifications

Date of Issuance: March 25, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 57

FACILITY OPERATING LICENSE NO. DPR-40

DOCKET NO. 50-285

1. Replace the following pages of the Operating License and the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
a) Operating License	
4	4
4a	-
5	
b) Appendix "A" Technical Specifications	
2-57c	2-57c
2-57d	2-57d
2-57e	2-57e
2-57f	-
5-20	5-20
-	5-21 (added)
6-2	6-2

2. Remove the following blank pages from the Appendix "A" Technical Specifications.

Remove

Figure 1-4
Figure 1-5
Figure 1-6
Figure 1-7
2-7a
2-55A
2-55B
2-55C
2-55D
2-55E

A. Maximum Power Level

Omaha Public Power District is authorized to operate the Fort Calhoun Station, Unit 1, at steady state reactor core power levels not to exceed 1500 megawatts thermal (rated power).

B. Technical Specifications

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

C. Security Plan

The licensee shall maintain in effect and fully implement all provisions of the Commission-approved physical security plan, including amendments and changes made pursuant to the authority of 10 CFR 50.54(p). The approved security plan consists of proprietary documents, collectively titled, "Fort Calhoun Station Unit No. 1, Site Security Plan," dated April 7, 1978, with Revision No. 1 dated July 31, 1978.

4. This amended license is effective as of the date of issuance and shall expire at midnight on June 7, 2008.

FOR THE ATOMIC ENERGY COMMISSION

Original signed by A. Giambusso

A. Giambusso, Deputy Director
for Reactor Projects
Directorate of Licensing

Enclosures:
Appendices A and B - Technical
Specifications

Date of Issuance: Aug 9, 1973

TECHNICAL SPECIFICATIONS

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Amendment No. 32, 34, 43, 54, 55, 57

2.0 LIMITING CONDITIONS FOR OPERATION

2.10 Reactor Core (Continued)

2.10.4 Power Distribution Limits (Continued)

(5) DNBR Margin During Power Operation Above 15% of Rated Power

(a) The following DNB related parameters shall be maintained within the limits shown:

(i)	Cold Leg Temperature	$\leq$ 545°F *
(ii)	Pressurizer Pressure	$\geq$ 2075 psia*
(iii)	Reactor Coolant Flow	$\geq$ 195,700 gpm**
(iv)	Axial Shape Index, Y_I	$\leq$ Figure 2-7

(b) With any of the above parameters exceeding the limit, restore the parameter to within its limit within 2 hours or reduce power to less than 15% of rated power within the next 8 hours.

Basis

Linear Heat Rate

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System, or the Incore Detector Monitoring System, provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring the axial shape index with the operable quadrant symmetric excore neutron flux detectors and verifying that the axial shape index is maintained within the allowable limits of Figure 2-6 as adjusted by Specification 2.10.4.(1).(c) for the allowed linear heat rate of Figure 2-5, RC Pump configuration, and $F_{xy}T$ of Figure 2-9. In conjunction with the use of the excore monitoring system and in establishing the axial shape index limits, the following assumptions are made: (1) the CEA insertion limits of Specification 2.10.2.(6) and long term insertion limits of Specification 2.10.2.(7) are satisfied, (2) the flux peaking augmentation factors are as shown in Figure 2-8, (3) the azimuthal power tilt restrictions of Specification 2.10.4.(4) are satisfied, and (4) the total planar radial peaking factor does not exceed the limits of Specification 2.10.4.(3).

*Limit not applicable during either a thermal power ramp in excess of 5% of rated thermal power per minute or a thermal power step of greater than 10% of rated thermal power.

**This number is an actual limit (not including uncertainties). All other values in this listing are indicated values and include an allowance for measurement uncertainty (e.g., 545°F, indicated, allows for an actual T_c of 547°F).

2.10 Reactor Core (Continued)

2.10.4 Power Distribution Limits (Continued)

The Incore Detector Monitoring System provides a direct measure of the peaking factors and the alarms which have been established for the individual incore detector segments ensure that the peak linear heat rates will be continuously maintained within the allowable limits of Figure 2-5. The setpoints for these alarms include allowances, set in the conservative directions, for the factors listed in 2.10.4.(1).

Total Planar and Integrated Radial Peaking Factors (F_{xy}^T and F_R^T) and Azimuthal Power Tilt (T_q)

The limitations of F_{xy}^T and T_q are provided to ensure that the assumptions used in the analysis for establishing the Linear Heat Rate and Local Power Density - High LCO's and LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. The limitations on F_R^T and T_q are provided to ensure that the assumptions used in the analysis establishing the DNB Margin LCO, and Thermal Margin/Low Pressure LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. If F_{xy}^T, F_R^T or T_q exceed their basic limitations, operation may continue under the additional restrictions imposed by the action statements since these additional restrictions provide adequate assurance that the assumptions used in establishing the Linear Heat Rate, Thermal Margin/Low Pressure and Local Power Density - High LCO's and LSSS setpoints remain valid. An azimuthal power tilt >0.10 is not expected and if it should occur, subsequent operation would be restricted to only those operations required to identify the cause of this unexpected tilt.

The value of T_q that must be used in the equation $F_{xy}^T = F_{xy}(1 + T_q)$ and $F_R^T = F_R(1 + T_q)$ is the measured tilt.

The surveillance requirements for verifying that F_{xy}^T, F_R^T, and T_q are within their limits provide assurance that the actual values of F_{xy}^T and T_q do not exceed the assumed values. Verifying F_{xy}^T and F_R^T after each fuel loading prior to exceeding 70% of rated power provides additional assurance that the core was properly loaded.

DNBR Margin During Power Operation Above 15% of Rated Power

The selection of limiting safety system settings and reactor operating limits is such that:

1. No specified acceptable fuel design limits will be exceeded as a result of the design basis anticipated operational occurrences, and

2. The consequences of the design basis postulated accidents will be no more severe than the predicted acceptable consequences of the accident analysis in Section 14.

In order for these objectives to be met, the reactor must be operated consistent with the operating limits specified for margin to DNB.

The parameter limits given in (5) and Figure 2-9 along with the parameter limits on quadrant tilt and control element assembly position (Figure 2-4) provide a high degree of assurance that the DNB overpower margin will be maintained during steady state operation.

The actions specified assure that the reactor is brought to a safe condition.

The reactor coolant pump differential pressure monitoring system that will be used to measure flow provides an accurate method of determining reactor coolant flow.

The procedure for determining individual pump and reactor vessel flow will be as follows:

1. Obtain a pump casing ΔP, using the precision resistor and high accuracy digital voltmeter and converting to pressure.

2. Obtain cold leg temperature and pressurizer pressure.

3. Correct the reading to the curve specific gravity.

4. Obtain pump flows from individual pump casing vs. flow curves.

5. Add the individual pump flows to obtain the best estimate reactor vessel flow.

5.0 ADMINISTRATIVE CONTROLS

5.12 Environmental Qualification

5.12.1 By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of: Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors" (DOR Guidelines); or, NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," December 1979. Copies of these documents are attached to Order for Modification of License DPR-40 dated October 24, 1980.

5.12.2 By no later than December 1, 1980, complete and auditable records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

5.13 Secondary Water Chemistry

A secondary water chemistry monitoring program to inhibit steam generator tube degradation shall be implemented. This program shall be described in the station chemistry manual and shall include:

1. Identification of a sampling schedule for the critical parameters and control points for these parameters;

2. Identification of the procedures used to measure the values of the critical parameters;

3. Identification of process sampling points;

4. Procedures for the recording and management of data;

5. Procedures defining corrective actions for off control point chemistry conditions; and

6. A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective actions.

5.14 Systems Integrity

A program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as low as practical levels shall be implemented. This program shall include the following:

1. Provisions establishing preventive maintenance and periodic visual inspection requirements, and

2. Integrated leak test requirements for each system at a frequency not to exceed refueling cycle intervals.

5.15 Iodine Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions shall be implemented. This program shall include the following:

1. Training of personnel,

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

6.0 INTERIM SPECIAL TECHNICAL SPECIFICATION

6.2 Use of Spent Fuel Shipping Cask

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