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. 39
License No. DPR-40

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

   A. The application for amendment by Omaha Public Power District (the licensee) dated August 5, 1977, 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, the license 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 Appendices A and B, as revised through Amendment No. 39, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 7, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 39

TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-40

DOCKET NO. 50-285

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised page is identified by Amendment number and contains vertical lines indicating the area of change.

Remove	Replace
2-2	2-2
-	2-2a (new)
2-15	2-15
2-16	2-16
2-22	2-22
2-23	2-23
-	2-23a (new)
3-16	3-16
-	3-16a (new)

(8) Maximum steam generator steam side leak test pressure shall not exceed 1000 psia. A minimum temperature of 82°F is required.

(9) A non-operating reactor coolant pump shall not be started unless at least one of the following conditions is met:

(a) A pressurizer steam space of 60% by volume or greater exists, or

(b) The steam generator secondary side temperature is less than 50°F above that of the reactor coolant system cold leg.

Basis

When reactor coolant boron concentration is being changed, the process must be uniform throughout the reactor coolant system volume to prevent stratification of reactor coolant at lower boron concentration which could result in a reactivity insertion. Sufficient mixing of the reactor coolant is assured if one low pressure safety injection pump or one reactor coolant pump is in operation. The low pressure safety injection pump will circulate the reactor coolant system volume in less than 35 minutes when operated at rated capacity. The pressurizer volume is relatively inactive; therefore, it will tend to have a boron concentration higher than the rest of the reactor coolant system during a dilution operation. Administrative procedures will provide for use of pressurizer sprays to maintain a nominal spread between the boron concentration in the pressurizer and the reactor coolant system during the addition of boron.(1)

Both steam generators are required to be filled above the low steam generator water level trip set point whenever the temperature of the reactor coolant is greater than the design temperature of the shutdown cooling system to assure a redundant heat removal system for the reactor.

The design cyclic transients for the reactor system are given in FSAR Section 4.2.2. In addition, the steam generators are designed for additional conditions listed in FSAR Section 4.3.4. Flooded and pressurized conditions on the steam side assure minimum tube sheet temperature differential during leak testing. The minimum temperature for pressurizing the steam generator steam side is 70°F.

Formation of a 60% steam space ensures that the resulting pressure increase would not result in an overpressurization, should a reactor coolant pump be started when the steam generator secondary side temperature is greater than that of the RCS cold leg.

2.0 LIMITING CONDITIONS FOR OPERATION
2.1 Reactor Coolant System (Continued)
2.1.1 Operable Components (Continued)

For the case in which no pressurizer steam space exists, limitation of the steam generator secondary side/RCS cold leg ΔT to 50°F ensures that a single low set point PORV would prevent an overpressurization due to actuation of a reactor coolant pump.

References

(1) FSAR Section 4.3.7

2.0 LIMITING CONDITIONS FOR OPERATION
2.1 Reactor Coolant System (Continued)

2.1.6 Pressurizer and Steam System Safety Valves

Applicability

Applies to the status of the pressurizer and steam system safety valves.

Objective

To specify minimum requirements pertaining to the pressurizer and steam system safety valves.

Specifications

To provide adequate overpressure protection for the reactor coolant system and steam system, the following safety valve requirements shall be met:

(1) The reactor shall not be made critical unless the two pressurizer safety valves are operable with their lift settings adjusted to ensure valve opening between 2500 psia and 2545 psia ±1%.[1]

(2) Whenever there is fuel in the reactor, and the reactor vessel head is installed, a minimum of one operable safety valve shall be installed on the pressurizer. However, when in at least the cold shutdown condition, safety valve nozzles may be open to containment atmosphere during performance of safety valve tests or maintenance to satisfy this specification.

(3) Whenever the reactor is in power operation, eight of the ten steam safety valves shall be operable with their lift settings between 1000 psia and 1050 psia with a tolerance of ±1% of the nominal nameplate set point values.[1]

(4) Both pressurizer power-operated relief valves (PORV's) shall be operable, at the low setpoint, whenever the cold leg temperature is less than 300°F. One PORV may be inoperable for up to 7 days, provide the remaining PORV is operable. If the above conditions of this paragraph cannot be met, the primary system must be depressurized and vented.

Basis

The highest reactor coolant system pressure reached in any of the accidents analyzed was 2480 psia and resulted from a complete loss of turbine generator load without simultaneous reactor trip while operating at 1500 MWt.[2] The reactor is assumed to trip on a "High Pressurizer Pressure" trip signal.

To determine the maximum steam flow, the only other pressure relieving system assumed operational is the steam system safety valves. Conservative values for all systems parameters, delay times and core moderator, coefficients are assumed. Overpressure protection is provided to portions of the reactor coolant system which are at the highest pressure considering pump head, flow pressure drops and elevation heads.

If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve lift pressure would be less than half the capacity of one safety valve. This specification, therefore, provides adequate defense against overpressurization when the reactor is subcritical.

Performance of certain calibration and maintenance procedures on safety valves requires removal from the pressurizer. Should a safety valve be removed, either operability of the other safety valve or maintenance of at least one nozzle open to atmosphere will assure that sufficient relief capacity is available. Use of plastic or other similar material to prevent the entry of foreign material into the open nozzle will not be construed to violate the "open to atmosphere" provision, since the presence of this material would not significantly restrict the discharge of reactor coolant.

The total relief capacity of the ten steam system safety valves is 6.54×10^6 lb/hr. At the rated power of 1420 MWt, a relief capacity of only 4.7×10^6 lb/hr is required to prevent overpressurization of the steam system on loss-of-load conditions and eight valves provide relieving capability of 4.976×10^6 lb/hr.(3)

Alignment of the power-operated relief valve low setpoint below 300°F provides sufficient margin, when used in conjunction with Technical Specification Sections 2.1.1 and 2.3. to prevent the design basis pressure transients from causing an overpressurization incident. Limitation of this requirement to scheduled cooldown ensures that, should emergency conditions dictate rapid cooldown of the reactor coolant system, inoperability of the low temperature overpressure protection system would not prove to be an inhibiting factor.

Removal of the reactor vessel head provides sufficient expansion volume to limit any of the design basis pressure transients. Thus, no additional relief capacity is required.

References

(1) Article 9 of the 1968 ASME Boiler and Pressure Vessel Code, Section III

(2) FSAR, Section 14.9

(3) FSAR, Sections 4.3.4, 4.3.9.5

(3) Whenever the reactor coolant system cold leg temperature is below 210°F and the reactor vessel head is installed, at least two (2) HPSI pump control switches shall be placed in pull-stop.

Whenever the reactor coolant system cold leg temperature is below 110°F and the reactor vessel head is installed, all three (3) HPSI pump control switches shall be placed in pull-stop.

In the event that no charging pumps are operable, a single HPSI pump may be taken from pull-stop and utilized for boric acid injection to the core.

Basis

The normal procedure for starting the reactor is to first heat the reactor coolant to near operating temperature by running the reactor coolant pumps. The reactor is then made critical by withdrawing CEA's and diluting boron in the reactor coolant. With this mode of start-up, the energy stored in the reactor coolant during the approach to criticality is substantially equal to that during power operation and therefore all engineered safety features and auxiliary cooling systems are required to be fully operable. During low power physics tests at low temperatures, there is a negligible amount of stored energy in the reactor coolant; therefore, an accident comparable in severity to the design basis accident is not possible and the engineered safeguards systems are not required.

The SIRW tank contains a minimum of 283,000 gallons of usable water containing 1900 ppm boron.[1] This is sufficient boron concentration to provide a shutdown margin of 5%, including allowances for uncertainties, with all control rods withdrawn and a new core at a temperature of 60°F.[2]

The limits for the safety injection tank pressure and volume assure the required amount of water injection during an accident and are based on values used for the accident analyses. The minimum 116.2 inch level corresponds to a volume of 825 ft^3 and the maximum 128.1 inch level corresponds to a volume of 895.5 ft^3.

Prior to the time the reactor is brought critical, the valving of the safety injection system must be checked for correct alignment and appropriate valves locked. Since the system is used for shutdown cooling, the valving will be changed and must be properly aligned prior to start-up of the reactor.

The operable status of the various systems and components is to be demonstrated by periodic tests. A large fraction of these tests will be performed while the reactor is operating in the power range.

If a component is found to be inoperable, it will be possible in most cases to effect repairs and restore the system to full operability within a relatively short time. For a single component to be inoperable does not negate the ability of the system to perform its function, but it reduces the redundancy provided in the reactor design and thereby limits the ability to tolerate additional equipment failures. To provide maximum assurance that the redundant component(s) will operate if required to do so, the redundant component(s) is to be tested prior to initiating repair of the inoperable component. If it develops that the inoperable component is not repaired within the specified allowable time period, or a second component in the same or related system is found to be inoperable, the reactor will initially be put in the hot shutdown condition to provide for reduction of cooling requirements after a postulated loss-of-coolant accident. This will also permit improved access for repairs in some cases. After a limited time in hot shutdown, if the malfunction(s) is not corrected, the reactor will be placed in the cold shutdown condition utilizing normal shutdown and cooldown procedures. In the cold shutdown condition, release of fission products or damage of the fuel elements is not considered possible.

The plant operating procedures will require immediate action to effect repairs of an inoperable component and therefore in most cases repairs will be completed in less than the specified allowable repair times. The limiting times to repair are intended to assure that operability of the component will be restored promptly and yet allow sufficient time to effect repairs using safe and proper procedures.

The requirement for core cooling in case of postulated loss-of-coolant accident while in the hot shutdown condition is significantly reduced below the requirements for a postulated loss-of-coolant accident during power operation. Putting the reactor in the hot shutdown condition reduces the consequences of a loss-of-coolant accident and also allows more free access to some of the engineered safeguards components in order to effect repairs.

Failure to complete repairs within 48 hours of going to the hot shutdown condition is considered indicative of a requirement for major maintenance and, therefore, in such a case, the reactor is to be put into the cold shutdown condition.

With respect to the core cooling function, there is functional redundancy over most of the range of break sizes.[3][4]

The LOCA analysis confirms adequate core cooling for the break spectrum up to and including the 32 inch double-ended break assuming the safety injection capability which most adversely affects accident consequences and are defined as follows. The entire contents of all four safety injection tanks are assumed to be available for emergency core cooling, but the contents of one of the tanks is assumed to be lost through the reactor coolant system. In addition, of the three high-pressure safety injection pumps and the two low-pressure safety injection pumps, for large break analysis it is assumed that two high pressure and one low pressure operate while only one of each type is assumed to operate in the small break analysis[5]; and also that 25% of their combined discharge rate is lost from the reactor coolant system out of the break. The transient hot spot fuel clad temperatures for the break sizes considered are shown on FSAR Figures 1-19 (Amendment No. 34).

Placing at least two HPSI pump control switches in pull-stop below 210°F results in no more than one HPSI pump remaining operable. A single low setpoint PORV is sufficient to prevent an overpressurization, caused by operation of one HPSI pump and three charging pumps, above a cold leg temperature of 110°F. Placing of all three HPSI pump control switches in pull-stop below 110°F results in no HPSI pumps remaining operable. A single low setpoint PORV is sufficient to prevent an overpressurization, caused by operation of three charging pumps, at any cold leg temperature.

Technical Specification 2.2(1) specifies that, when fuel is in the reactor, at least one flow path shall be provided for boric acid injection to the core. Should boric acid injection become necessary, and no charging pumps are operable, operation of a single HPSI pump would provide the required flow path.

References

(1) FSAR, Section 14.15.1

(2) FSAR, Section 6.2.3.1

(3) FSAR, Section 14.15.3

(4) FSAR, Appendix K

(5) Omaha Public Power District's Submittal, December 1, 1976.

TABLE 3-3 (Continued)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF MISCELLANEOUS INSTRUMENTATION AND CONTROLS

	Channel Description	Surveillance Function	Frequency	Surveillance Method
14.	Nuclear Detector Well Cooling Annulus Exit Air Temperature Detectors	a. Test	S	a. Compare eight (8) independent readings.
		b. Calibrate	R	b. Calibrate with known temperature.
15.	Reactor Coolant System Flow	a. Check	M	a. Calculation of reactor coolant flow rate.
16.	Pressurizer Pressure	a. Check	S	a. Comparison of independent pressure readings.
17.	Reactor Coolant Inlet Temperature	a. Check	S	a. Comparison of independent temperature readings.
18.	Low-Temperature Setpoint Power-Operated Relief Valves	a. Test	PM	a. Verify operability of actuation circuitry for low-temperature setpoint power-operated relief valves by utilization of installed test switches.
		b. Calibrate	R	b. Calibrate temperature and pressure channels.

TABLE 3-3 (Continued)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING
OF MISCELLANEOUS INSTRUMENTATION AND CONTROLS

Channel Description	Surveillance Function	Frequency	Surveillance Method

S - Each Shift
D - Daily
M - Monthly
A - Annually
R - 18 Months
P - Prior to each startup if not performed within previous week.
PM - Prior to scheduled cold leg cooldown below 300°F; monthly whenever temperature remains below 300°F and reactor vessel head is installed.