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. 49
License No. DPR-40

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

   A. The applications for amendment by Omaha Public Power District (the licensee) dated July 5, 1979 and May 12, 1980, as supplemented May 14, 1980 comply 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.

8008060141

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:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 49, 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

Robert A. Clark

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 25, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 49

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 pages are identified by Amendment number and contain vertical lines indicating the area of change.

Pages

2-21

2-23

2-23a*

2-25

2-28

3-62

3-62a (added)

* Overleaf page provided for your convenience.

(2) Modification of Minimum Requirements

During power operation, the Minimum Requirements may be modified to allow one of the following conditions to be true at any one time. If the system is not restored to meet the minimum requirements within the time period specified below, the reactor shall be placed in a hot shutdown condition within 12 hours. If the minimum requirements are not met within an additional 48 hours the reactor shall be placed in a cold shutdown condition within 24 hours.

a. One low-pressure safety injection pump may be inoperable provided the pump is restored to operable status within 24 hours.

b. One high-pressure safety injection pump may be inoperable provided the pump is restored to operable status within 24 hours.

c. One shutdown heat exchanger and two of four component cooling water heat exchangers may be inoperable for a period of no more than 24 hours.

d. Any valves, interlocks or piping directly associated with one of the above components and required to function during accident conditions shall be deemed to be part of that component and shall meet the same requirements as listed for that component.

e. Any valve, interlock or piping associated with the safety injection and shutdown cooling system which is not covered under d. above but which is required to function during accident conditions may be inoperable for a period of no more than 24 hours.

f. One safety injection tank may be inoperable for a period of no more than one hour.

g. Level and pressure instrumentation on one safety injection tank may be inoperable for a period of one hour.

2.0 LIMITING CONDITIONS FOR OPERATION

2.3 Emergency Core Cooling System (Continued)

used for shut down 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. 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. If 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).

Inadvertent actuation of three (3) HPSI pumps and three (3) charging pumps, coincident with the opening of one of the two PORV's, would result in a peak primary system pressure of 1190 psia. 1190 psia corresponds with a minimum permissible temperature of 320°F on Figure 2-1B. Thus, at least one HPSI pump is disabled at 320°F.

Inadvertent actuation of two (2) HPSI pumps and three (3) charging pumps, coincident with the opening of one of the two PORV's, would result in a peak primary system pressure of 1040 psia. 1040 psia corresponds with a minimum permissible temperature of 310°F on Figure 2-1B. Thus, at least two HPSI pumps will be disabled at 310°F.

Inadvertent actuation of one (1) HPSI and three (3) charging pumps, coincident with opening of one of the two PORV's, would result in a peak primary system pressure of 685 psia. 685 psia corresponds with a minimum allowable temperature of 276°F on Figure 2-1B. Thus, all three HPSI pumps will be disabled at 276°F.

Inadvertent actuation of three (3) charging pumps, coincident with opening of one of the two PORV's, would result in a peak primary system pressure of 160 psia. 160 psia corresponds with a minimum allowable temperature of 78°F (approximately the boltup temperature of 82°F) on Figure 2-1B. Thus, disabling of the charging pumps is not required.

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

During power operation one of the components listed above (in addition to one raw water pump) may be inoperable. If the inoperable component is not restored to operability within seven days, the reactor shall be placed in a hot shutdown condition within 12 hours. If the inoperable component is not restored to operability within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours.

(2) Modification of Minimum Requirements

During power operation, the minimum requirements may be modified to allow a total of two of the components listed in a. and b. to be inoperable at any one time (in addition to one raw water pump) provided that the emergency diesel-generator connected to the other engineered safeguards 4.16-kV bus (1A4 or 1A3) is started to demonstrate operability. Only two raw water pumps may be out of service. If the operability of both components is not restored within 24 hours, the reactor shall be placed in a hot shutdown condition within 12 hours. If the operability of both components is not restored within an additional 48 hours, the reactor shall be placed in a cold shutdown condition within 24 hours.

Any valves, interlocks and piping directly associated with one of the above components and required to function during accident conditions shall be deemed to be part of that component and shall meet the same requirements as for that component.

Any valve, interlock or piping associated with the containment cooling system which is not included in the above paragraph and which is required to function during accident conditions may be inoperable for a period of no more than 24 hours. If operation is not restored within 24 hours, the reactor shall be placed in a hot shutdown condition within 12 hours.

Basis

The requirements of Section 2.3, Emergency Core Cooling System, apply to the specifications above with respect to the operability of the

2.0 LIMITING CONDITIONS FOR OPERATION

2.5 Steam and Feedwater Systems

Applicability

Applies to the operating status of the steam and feedwater systems.

Objective

To define certain conditions of the steam and feedwater system necessary to assure adequate decay heat removal.

Specifications

The reactor coolant shall not be heated about 300°F unless the following conditions are met:

(1) Both auxiliary feedwater pumps are operable. One of the auxiliary feedwater pumps may be inoperable for 24 hours provided that the redundant component shall be tested to demonstrate operability.

(2) A minimum of 55,000 gallons of water in the emergency feedwater storage tank and a backup water supply to the emergency feedwater storage tank from the Missouri River by the fire water system.

(3) All valves, interlocks and piping associated with the above components required to function during accident conditions are operable. Manual valves that could interrupt auxiliary feedwater flow to the steam generators shall be locked in the required position to ensure a flow path to the steam generators.

(4) The main steam stop valves are operable and capable of closing in four seconds or less under no-flow conditions.

Basis

A reactor shutdown from power requires a removal of core decay heat. Immediate decay heat removal requirements are normally satisfied by the steam bypass to the condenser. Therefore, core decay heat can be continuously dissipated via the steam bypass to the condenser as long as feedwater to the steam generator is available. Normally, the capability to supply feedwater to the steam generators is provided by operation of the turbine cycle feedwater system. In the unlikely event of complete loss of electrical power to the station, decay heat removal is by steam discharge to the atmosphere via the main steam safety and atmospheric dump valves. Either auxiliary feedwater pump can supply sufficient feedwater for removal of decay heat from the plant. The minimum amount of water in the emergency feedwater storage tank is the amount needed for 8 hours of such operation. The tank can be resupplied with water from the fire protection system.[1]

3.0 SURVEILLANCE REQUIREMENTS

3.9 Auxiliary Feedwater System

Applicability

Applies to periodic testing requirements of the turbine-driven and motor-driven auxiliary feedwater pumps.

Objective

To verify the operability of the auxiliary feedwater (AFW) system and its ability to respond properly when required.

Specifications

(1) The position of valves necessary to ensure auxiliary feedwater flow to the steam generators shall be verified by a monthly inspection. Anytime maintenance is performed on the auxiliary feedwater system which alters valve alignments, an operator shall check that the AFW system valves are properly aligned, to ensure AFW flow to the steam generators, and a second operator shall independently verify proper valve alignment.

(2) The operability of the motor-driven auxiliary feedwater pump, the steam turbine-driven auxiliary feedwater pump, and the auxiliary feedwater pumps' steam generator level regulating valves HCV-1107A, HCV-1107B, HCV-1108A, HCV-1108B, and auxiliary feedwater cross-tie valve HCV-1384 shall be confirmed at least every three months.

(3) The capabilities of the motor-driven and turbine-driven auxiliary feedwater pumps shall be verified by using local pressure indicators and flow indicators in the control room. The discharge pressure will be verified to be 40 psig above the steam generator pressure at rated steam flow.

(4) Following cold shutdown and prior to raising the reactor coolant temperature above 300°F, the motor-driven auxiliary feedwater pump shall be tested to verify the normal flow path for auxiliary feedwater to the steam generators.

Basis

The valve position verifications performed monthly and following auxiliary feedwater system maintenance will confirm the availability of an auxiliary feedwater flow path to the steam generators.

3.0 SURVEILLANCE REQUIREMENTS

3.9 Auxiliary Feedwater System

The testing every three months and after cold shutdowns of the auxiliary feedwater pumps will verify their operability by recirculating water to the emergency feedwater storage tank and operating, one at a time, the regulating valves (HCV-1107B and HCV-1108B) to confirm a flow path to the steam generators and operability of the valves.

Proper functioning of the steam turbine admission valve and starting of the feedwater pump will demonstrate the integrity of the steam driven pump. Verification of correct operation will be made both from instrumentation within the main control room and direct visual observation of the pumps.

References

(1) FSAR, Section 9.4

(2) Technical Specification 2.5