



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. 60  
License No. DPR-40

1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The applications for amendment by the Omaha Public Power District (the licensee) dated March 13, 1978 and May 29, 1981, 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.

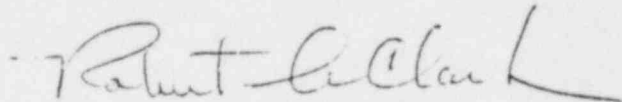
2. Accordingly, Facility Operating License No. DPR-40 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Facility Operating License No. DPR-40 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 60, 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, Chief  
Operating Reactors Branch #3  
Division of Licensing

Attachment:  
Changes to the Technical  
Specifications

Date of Issuance: July 29, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 60

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.

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

tube rupture accident. (3) The potential dose at the site boundary for this accident is larger and hence more limiting than the dose that would result from one year of operation with the maximum reactor coolant activity combined with the maximum permissible unidentified leakage from the reactor coolant system (Section 2.1.4 of these Specifications).

The accidents for which primary and secondary coolant concentrations become the limiting parameters are a steam generator tube rupture and a steam line break upstream of the isolation valves, respectively, with the following basic assumptions:

(1) Steam Generator Tube Rupture

- a. Maximum permissible two-hour doses at the exclusion distance of 1.5 rem to the thyroid, and 0.5 rem whole body.
- b. Loss of offsite power.
- c. 90,000 lbs. of primary coolant is released to the secondary.
- d. Identification of the accident and pressure equalization between primary and secondary occurs within 30 minutes.
- e. Ten percent of the primary coolant entering the secondary system through the ruptured tube flashes and is released directly to the environs.
- f. The quantity of steam released from the secondary system is determined from mass and energy balances on the primary and secondary systems.
- g. Partitioning of iodine in the steam released from both steam generators by a factor of 0.1.

(2) Steam Line Break

- a. Maximum permissible two-hour doses at the exclusion distance of 1.5 rem to the thyroid, and 0.5 rem whole body.
- b. Loss of offsite power.
- c. Blowdown of the contents of one steam generator.

## 2.0 LIMITING CONDITIONS FOR OPERATION

### 2.7 Electrical Systems (Continued)

- g. One of the four a-c instrument buses may be inoperable for 8 hours provided the reactor protective and engineered safeguards systems instrument channels supplied by the remaining three buses are all operable.
- h. Two battery chargers may be inoperable for up to 8 hours provided battery charger No. 1 or No. 2 is operable.
- i. Either one of the diesel generators may be inoperable for up to seven days (total for both) during any month, provided the other diesel is started to verify operability, shutdown and controls are left in the automatic mode and there are no inoperable engineered safeguards components associated with the operable diesel generator.
- j. Island buses 1B3A-4A, 1B3B-4B, and 1B3C-4C may be inoperable for up to 8 hours provided there are no inoperable safeguards components associated with the operable bus which are redundant to the inoperable bus.
- k. Either one of the DC buses (Panels EE-8F and EE-8G) may be inoperable for up to 8 hours.
- l. Either one of the DC Distribution Panels AI-41A and AI-41B may be inoperable for up to 8 hours.
- m. AC Instrument Panel AI-42A or AI-42B may be inoperable for up to 8 hours.
- n. The 161 kV transmission line may be out of service and unit operation may continue or the reactor may be restarted from a hot shutdown condition if (i) operability of the remaining source is immediately verified and (ii) immediate notification is made by telephone or telegraph to the Director of the NRC Regional Office in Arlington, Texas of the loss and of the plans to restore the electric power system to its full capability.

#### Basis

The electrical system equipment is arranged so that no single failure can inactivate enough engineered safeguards to jeopardize the plant safety. The 480 V safeguards are arranged on nine bus sections. The 4.16 kV safeguards are supplied from two buses.

The normal source of auxiliary power with the plant at power for the safeguards buses is from the house service power transformers being fed from the 161 kV incoming line with on-site emergency power

TABLE 2-2

Instrument Operating Requirements for Reactor Protective System

<u>No.</u>	<u>Functional Unit</u>	<u>Minimum Operable Channels</u>	<u>Minimum Degree of Redundancy</u>	<u>Permissible Bypass Condition</u>
1	Manual (Trip Buttons)	1	None	None
2	High Power Level	2(bc)	1(c)	Thermal Power Input Bypassed below $10^{-4}\%$ of Rated Power (a) (d)
3	Thermal Margin/Low Pressurizer Pressure	2(b)	1	Below $10^{-4}\%$ of Rated Power (a) (d)
4	High Pressurizer Pressure	2(b)	1	None
5	Low R.C. Flow	2(b)	1	Below $10^{-4}\%$ of Rated Power (a) (d)
6	Low Steam Generator Water Level	2/Steam Gen (b)	1/Steam Gen	None
7	Low Steam Generator Pressure	2/Steam Gen (b)	1/Steam Gen	Below 550 psia (a) (d)
8	Containment High Pressure	2(b)	1	During Leak Test
9	Axial Power Distribution	2(bc)	1(c)	Below 15% of Rated Power (a)
10	High Rate Trip-Wide Range Log Channels	2	1	Below $10^{-4}\%$ and above 15% of rated power (a) (e)
11	Loss of Load	2(b)	1	Below 15% of rated power (a)

a Bypass automatically removed.

b One of the inoperable channels must be in the tripped condition.

c If two channels are inoperable, load shall be reduced to 70% or less of rated power.

d For low power physics testing this trip may be bypassed up to  $10^{-1}\%$  of rated power.

e For each channel, the same bistable automatically activates the Loss of Load and Axial Power Distribution (APD) trips and automatically bypasses the high rate trip at 15% of rated power. Only the APD trip is a Limiting Safety System Setting. Therefore, the bistable is set to actuate within the APD tolerance band.

TABLE 2-7

FIRE DETECTION ZONES

<u>Zone No.</u>	<u>Location</u>
1	Auxiliary Building, Elevations 971 and 989 West
2	Auxiliary Building, Elevation 989 East
3	Auxiliary Building, Elevation 989, Lower Electrical Penetration Room (Room 20)
4	Auxiliary Building, Elevation 989, Air Compressor Room (Room 19)
5	Auxiliary Building, Elevation 1007, Corridor 26, Rooms 58, 59, and 60
6	Auxiliary Building, Elevations 1007 and 1011, Uncontrolled
7	Auxiliary Building, Elevation 1013, Upper Electrical Penetration Room (Room 57)
8	Auxiliary Building, Elevations 989 and 1007, Boric Acid Tank Area, Drumming Area, New Fuel Area
9	Auxiliary Building, Elevation 1036, Control Room Complex, Control Room Hallways
10	Containment, Elevation 1013, RC Pump Cavities
11	Containment, Elevation 994
12	Containment, Elevation 1045
13	Auxiliary Building, Elevation 1025 (Rooms 69 and 71)
14	Turbine Building, Elevation 990
15	Turbine Building, Elevation 1011
16	Turbine Building, Elevation 1036
17	Containment Fans VA-3B and VA-7D
18	Containment Fans VA-3A and VA-7C
19	Containment Fans VA-2A and VA-2B
20	Control Room Panels CB-1/2/3 Return Air
21	Containment NDWC Fans VA-12A and VA-12B
22	Containment Purge Discharge Fans VA-32A and VA-32B
23	DG-2 Room Exhaust Fan, VA-52B
24	Containment Purge Supply Fans VA-24A and VA-24B
25	Control Room and Hallway Ventilation Ducts
26	Auxiliary Building (Controlled) Supply Fans, VA-35A and VA-35B
27	Auxiliary Building (Controlled) Exhaust Fans, VA-40A, VA-40B, and VA-40C
28	Auxiliary Building (Uncontrolled) Supply Fans, VA-45A and VA-45B
29	Auxiliary Building (Uncontrolled) Exhaust Fan, VA-41
30	Auxiliary Building Elevator Shaft Fan, VA-51
31	Control Room Air Conditioning Fans, VA-46A and VA-46B
32	DG-1 Room Exhaust Fan, VA-52A
33	Auxiliary Building, Elevation 1036 (Room 81)
34	Plant Sprinkler Flow
35	Auxiliary Building, DG-2 (Room 64)
36	Auxiliary Building, DG-1 (Room 63)
37	Intake Structure Including Raw Water Pump Room
38	Auxiliary Building Open Stairwell
39	Auxiliary Building Open Hatchway



TABLE 2-7  
(Continued)

HALON AREA FIRE ZONES

Zone  
No.

1	Cable Spreading Room
2	Cable Spreading Room
3	Control Room Walk-In Cabinets
4	Control Room Walk-In Cabinets
5	Switchgear Room - West
6	Switchgear Room - West
7	Switchgear Room - East
8	Switchgear Room - East

TABLE 2-8

FIRE HOSE STATION LOCATIONS

<u>No.</u>	<u>Location</u>	<u>Elevation</u>	<u>Size</u>	
1.	FP-4N	Intake Structure	1012'-6"	1.5"/2.5"
2.	FP-4P	Intake Structure	1012'-6"	1.5"/2.5"
3.	FP-3C	Yard Area	At grade level	2.5"
4.	FP-3B	Yard Area	At grade level	2.5"
5.	FP-3A	Yard Area	At grade level	2.5"
6.	FP-3F	Yard Area	At grade level	2.5"
7.	FP-3E	Yard Area	At grade level	2.5"
8.	FP-3D	Yard Area	At grade level	2.5"
9.	FP-7A	Auxiliary Building	989'-0"	1.5"/2.5"
10.	FP-7B	Auxiliary Building	989'-0"	1.5"/2.5"
11.	FP-7C	Auxiliary Building	989'-0"	1.5"/2.5"
12.	FP-7D	Auxiliary Building	989'-0"	1.5"/2.5"
13.	FP-7E	Auxiliary Building	989'-0"	1.5"/2.5"
14.	FP-7F	Auxiliary Building	989'-0"	1.5"/2.5"
15.	FP-7G	Auxiliary Building	989'-0"	1.5"/2.5"
16.	FP-8A	Auxiliary Building	1011'-0"	1.5"/2.5"
17.	FP-8B	Auxiliary Building	1011'-0"	1.5"/2.5"
18.	FP-8C	Auxiliary Building	1011'-0"	1.5"/2.5"
19.	FP-8D	Auxiliary Building	1007'-6"	1.5"/2.5"
20.	FP-8E	Auxiliary Building	1007'-6"	1.5"/2.5"
21.	FP-8F	Auxiliary Building	1007'-6"	1.5"/2.5"
22.	FP-8G	Auxiliary Building	1007'-6"	1.5"/2.5"
23.	FP-8H	Auxiliary Building	1007'-6"	1.5"/2.5"

TABLE 2-8 (Continued)

FIRE HOSE STATION LOCATIONS

<u>No.</u>	<u>Location</u>	<u>Elevation</u>	<u>Size</u>	
24.	FP-9A	Auxiliary Building	1025'-0"	1.5"/2.5"
25.	FP-9B	Auxiliary Building	1025'-0"	1.5"/2.5"
26.	FP-9C	Auxiliary Building	1025'-0"	1.5"/2.5"
27.	FP-9D	Auxiliary Building	1025'-0"	1.5"/2.5"
28.	FP-10A	Auxiliary Building	1036'-0"	1.5"/2.5"
29.	FP-10B	Auxiliary Building	1036'-0"	1.5"/2.5"
30.	FP-10C	Auxiliary Building	1036'-0"	1.5"/2.5"
31.	FP-10D	Auxiliary Building	1036'-0"	1.5"/2.5"
32.	FP-10E	Auxiliary Building	1036'-0"	1.5"/2.5"

TABLE 3-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TESTING OF REACTOR PROTECTIVE SYSTEM

	<u>Channel Description</u>	<u>Surveillance Function</u>	<u>Frequency</u>	<u>Surveillance Method</u>
1.	Power Range Safety Channels	a. Check	S	a. Comparison of four power channel readings, for both neutron flux and thermal power.
		b. Adjustment	D <sup>(4)</sup>	b. Channel adjustment to agree with heat balance calculation.
		c. Calibrate and Test	M <sup>(2)</sup>	c. Internal test signal to verify trips, alarms, permissives and actioner circuits.
2.	Wide-Range Logarithmic Neutron Monitors	a. Check	S	a. Comparison of four wide-range readings.
		b. Test <sup>(3)</sup>	P	b. Internal test signals to verify SUR indication and trip, power level permissives, instrument accuracy.
3.	Reactor Coolant Flow	a. Check	S	a. Comparison of four separate total flow indications.
		b. Calibrate	R	b. Known differential pressure applied to sensors to calibrate all loop devices.
		c. Test	M <sup>(2)</sup>	c. Bistable trip tester. <sup>(1)</sup>

TABLE 3-5  
(Continued)

	Test	Frequency	FSAR Section Reference
10c.	(Continued)	4. Automatic and/or Manual initiation of the system shall be demonstrated.	At least once per plant operating cycle.
11.	Containment Cooling and Iodine Removal Fuseable Linked Dampers	1. Demonstrate damper action. 2. Test a spare fuseable link.	1 year, 2 years, 5 years, and every 5 years thereafter 5.10
12.	Fuel Elements	Visually inspect fuel elements removed from the reactor.	During each refueling outage 3
13.	Diesel Generator Under-Voltage Relays	Calibrate	During each refueling outage 8.4.3
14.	Motor Operated Safety Injection Loop Valve Motor Starters (ICV-11, 314, 317, 320, 327, 329, 331, 333, 312, 315, 318, 321)	Verify the contactor pickup value at $\leq 85\%$ of 460 V.	During each refueling outage
15.	Pressurizer Heaters	Verify control circuits operation for post-accident heater use.	During each refueling outage.

4. DESIGN FEATURES

4.3 Nuclear Steam Supply System (Continued)

4.3.1 Reactor Coolant System (Continued)

The reactor coolant system is designed and constructed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Vessels including all addenda through the winter of 1967 and the Code for Pressure Piping USAS B31.1. The reactor coolant system is designed for a pressure of 2500 psia and a temperature of 650°F except for the pressurizer which has a design temperature of 700°F. The volume of the reactor coolant system is approximately 6,616 cubic feet.

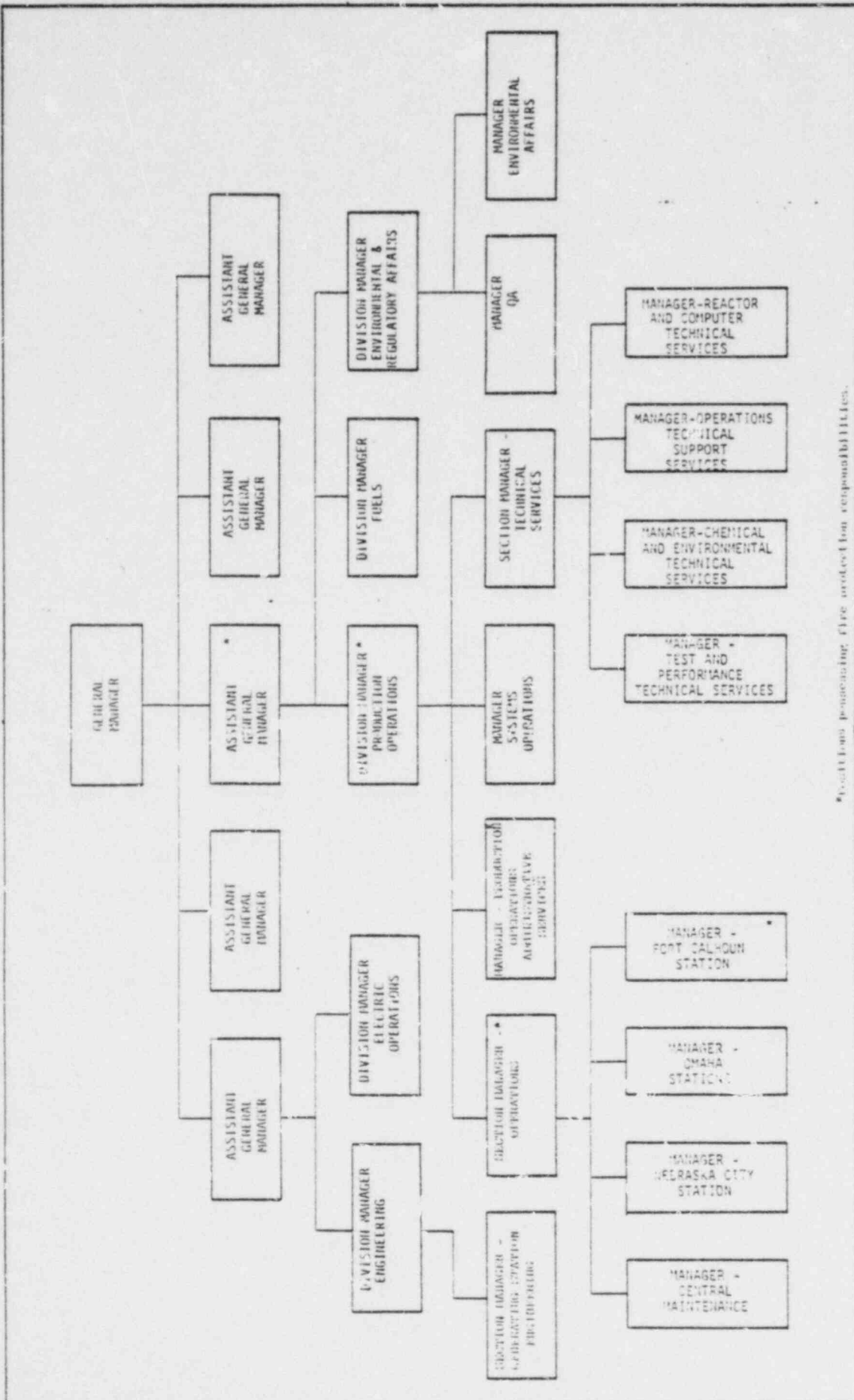
4.3.2 Reactor Core and Control

The reactor core shall approximate a right circular cylinder with an equivalent diameter of 106.5 inches and an active height of 128 inches. The reactor core shall normally consist of Dircaloy-4 clad fuel rods containing slightly enriched uranium in the form of sintered  $UO_2$  pellets. The fuel rods shall normally be grouped into 133 assemblies.

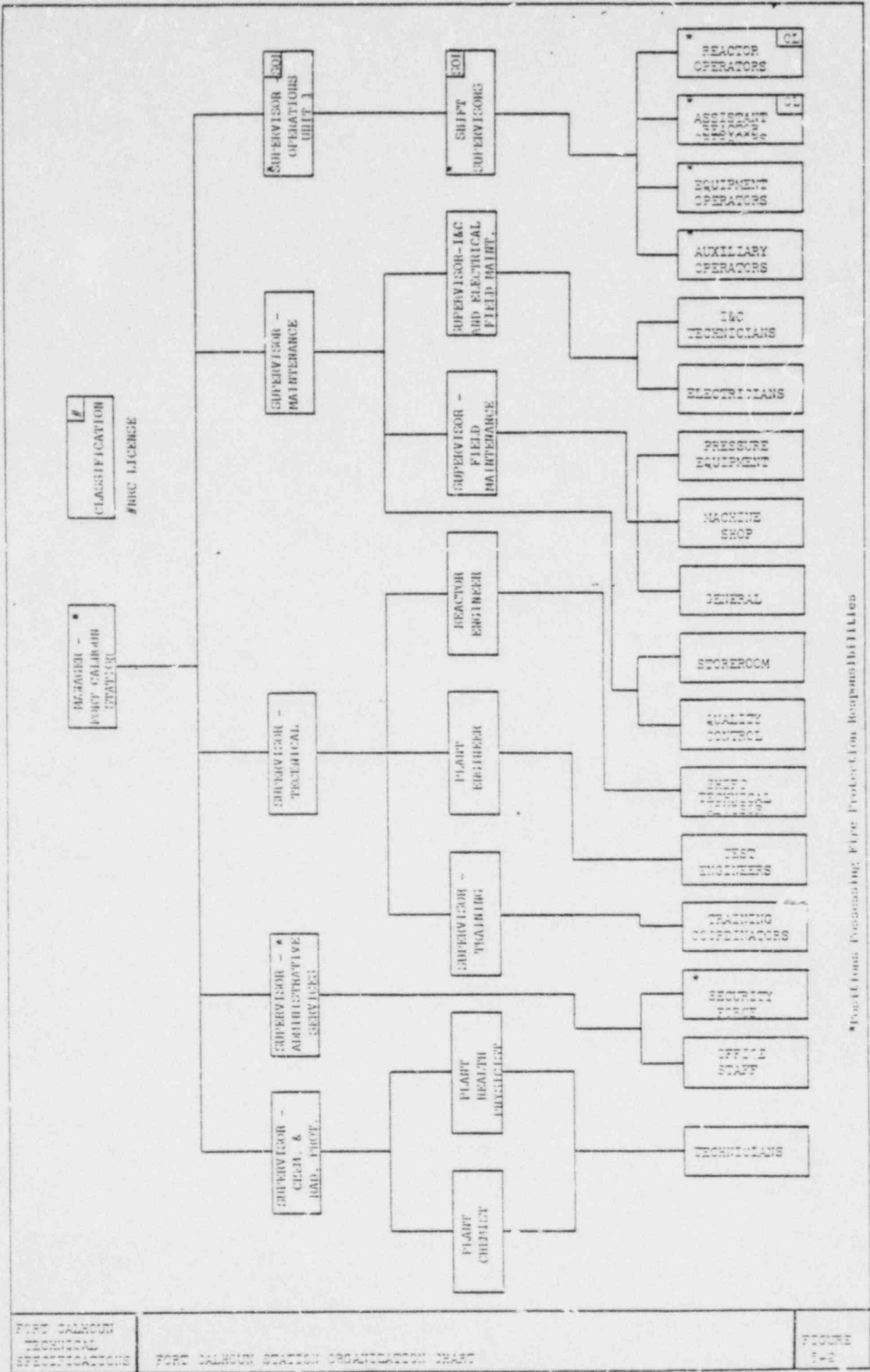
The core excess reactivity shall be controlled by a combination of boric acid chemical shim, control element assemblies, and mechanically fixed boron rods where required. Forty-nine control element assemblies are distributed throughout the core as shown in Figure 3.4-5 of the FSAR; four of the CEAs contain part-length adsorbers.

4.3.3 Emergency Core Cooling

Emergency core cooling is provided by the Safety Injection System which consists of various subsystems, each with internal redundancy. Included in the Safety Injection System are four safety injection tanks, three high-pressure and two low-pressure safety injection pumps, a safety injection and refueling water storage tank, and interconnecting piping as shown in FSAR Section 6.



\*Positions possessing fire protection responsibilities.



\* Positions Reserving Fire Protection Responsibilities



5.0 ADMINISTRATIVE CONTROLS

- 5.5.1.7 b. Render determinations in writing with regard to whether or not each item considered under 5.5.1.6(a) through (e) above constitutes an unreviewed safety question.
- c. Provide immediate written notification to the Section Manager - Operations and the Safety Audit and Review Committee of disagreement between the Plant Review Committee and the Manager - Fort Calhoun Station; however, the Manager - Fort Calhoun Station shall have responsibility for resolution of such disagreements pursuant to 5.2.1 above.

Records

- 5.5.1.8 The Plant Review Committee shall maintain written minutes of each meeting and copies shall be provided to the Section Manager - Operations and Chairman of the Safety Audit and Review Committee.

5.5.2 Safety Audit and Review Committee (SARC)

Function

- 5.5.2.1 The Safety Audit and Review Committee shall function to provide the independent review and audit of designated activities in the areas of:
- a. nuclear power plant operation;
  - b. nuclear engineering
  - c. chemistry and radiochemistry
  - d. metallurgy
  - e. instrumentation and control
  - f. radiological safety
  - g. mechanical and electrical engineering
  - h. quality assurance

Composition

- 5.5.2.4 The Safety Audit and Review Committee shall be composed of:

Chairman: Division Manager - Environmental and Regulatory Affairs

Member: Assistant General Manager - Production Operations, Fuels, and ERA

Member: Assistant General Manager - Electric Operations and Engineering

Member: Division Manager - Engineering

Member: Division Manager - Production Operations

Member: OPPD Operations, Engineering, and Technical Support Staff

Member: Qualified Non-District Affiliated Consultants as Required and as Determined by SARC Chairman

5.0 ADMINISTRATIVE CONTROLS

Alternates

- 5.5.2.3 Alternate members shall be appointed in writing by the Chairman of the Safety Audit and Review Committee to serve on a temporary basis; however, no more than two alternates may participate in the Safety Audit and Review Committee activities at any one time.

Consultants

- 5.5.2.4 Consultants shall be utilized as determined by the Safety Audit and Review Committee Chairman to provide expert advice to the Safety Audit and Review Committee.

Meeting Frequency

- 5.5.2.5 The Safety Audit and Review Committee shall meet at least once every six months.

Quorum

- 5.5.2.6 A quorum of the Safety Audit and Review Committee shall consist of the Chairman or his designated alternate and a majority of the Safety Audit and Review Committee members including alternates. No more than a minority of the quorum shall have line responsibility for the operation of the nuclear plant.

Review

- 5.5.2.7 The Safety Audit and Review Committee shall review:
- a. The safety evaluations for 1) procedures, equipment or systems and 2) tests or experiments completed under the provision of section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.
  - b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in section 50.59, 10 CFR.

## 5.0 ADMINISTRATIVE CONTROLS

- 5.5.2.7 c. Proposed tests or experiments which involve an unreviewed safety question as defined in section 50.59, 10 CFR.
- d. Proposed changes in Technical Specifications or licenses.
- e. Violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.
- f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.
- g. All events which are required by regulations or Technical Specifications to be reported to the NRC within 24 hours.
- h. Any indication of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.
- i. Reports and meeting minutes of the Plant Review Committee.

The Chairman of the Safety Audit and Review Committee (SARC) may designate subgroups, special working committees, or audit teams as he deems necessary in order to carry out the responsibilities of the SARC. These subgroups, committees, or audit teams will perform the SARC responsibilities and report on their activities for review at the next regularly scheduled SARC meeting following any group's action.

### Audit

- 5.5.2.8 Audits of facility activities shall be performed under the cognizance of the Safety Audit and Review Committee. These audits shall encompass:
- a. The conformance of facility operation to all provisions contained within the Technical Specifications and applicable license conditions at least once per year.
- b. The performance, training and qualifications of the entire facility staff at least once per year.
- c. The results of all actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per six months.
- d. The performance of all activities required by the Quality Assurance Program to meet the criteria of Appendix B, 10 CFR 50, at least once per two years.

## 5.0 ADMINISTRATIVE CONTROLS

- 5.5.2.3 e. The Fort Calhoun Station Emergency Plan and implementing procedures at least once per two years.
- f. The Site Security Plan and implementing procedures at least once per two years.
- g. Any other area of facility operation considered appropriate by the Safety Audit and Review Committee or the Assistant General Manager - Production Operations, Fuels, and Environmental & Regulatory Affairs.

### Authority

- 5.5.2.9 The Safety Audit and Review Committee shall report to and advise the Assistant General Manager - Production Operations, Fuels, and Environmental & Regulatory Affairs on those areas of responsibility specified in Section 5.5.2.7 and 5.5.2.8.

### Records

- 5.5.2.10 Records of Safety Audit and Review Committee activities shall be prepared, approved and distributed as indicated below:
- a. Minutes of each Safety Audit and Review Committee meeting shall be prepared, approved and forwarded to the Assistant General Manager - Production Operations, Fuels, and Environmental & Regulatory Affairs within 14 days following each meeting.
- b. Reports of reviews encompassed by Section 5.5.2.7 e, f, g, and h above, shall be prepared, approved and forwarded to the Assistant General Manager - Production Operations, Fuels, and Environmental & Regulatory Affairs within 14 days following completion of the review.
- c. Audit reports encompassed by Section 5.5.2.3 above, shall be forwarded to the Assistant General Manager - Production Operations, Fuels, and Environmental & Regulatory Affairs and to the responsible management positions designated by the Safety Audit and Review Committee within 30 days after completion of the audit.

## 5.5.3 Fire Protection Inspection

- a. An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified off-site licensee personnel or an outside fire protection firm. The audit and inspection program responsibility shall rest with the Safety Audit and Review Committee.
- b. An inspection and audit of the fire protection and loss prevention program by an outside qualified fire consultant shall be performed at intervals no greater than 3 years.

## 5.6 Reportable Occurrence Action

- 5.6.1 The following actions shall be taken in the event of a REPORTABLE OCCURRENCE:
- a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 5.9.