TABLE OF CONTENTS (Continued)

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.

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 Compliance 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 2-9

DNB MARGIN

SPECIFIED OPERATING LIMITS

Monitored Parameter	OPERATING LIMIT 4 Pump
Cold Leg Temperature	≤(545°F)*
Pressurizer Pressure	≥(2075 psia)*
Reactor Coolant Flow	≥(195,700 gpm)**
Axial Shape Index	≤(Figure 2-7)

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

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

TABLE 2-2

Instrument Operating Requirements for Reactor Protective System

No.	Functional Unit	Minimum Operable Channels	Minimum Degree of Redundancy	Permissible Bypass Condition
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(e)
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(e)

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

Zone No.	Location
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

	No.	Location	Elevation	Size
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

	No.	Location	Elevation	Size
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

	Channel Description	Surveillance Function	Frequency	Surveillance Method
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(4)	b. Channel adjustment to agree with heat balance calculation.
		c. Calibrate and Test	M(2)	c. Internal test signal to verify trips, alarms, permissives and auctioneer circuits.
2.	Wide-Range Logarithmic Neutron Monitors	a. Check	S	a. Comparison of four wide-range readings.
		b. Test(3)	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(2)	c. Bistable trip tester.(1)

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	9.10
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 (HCV-311, 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.	

4.0 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

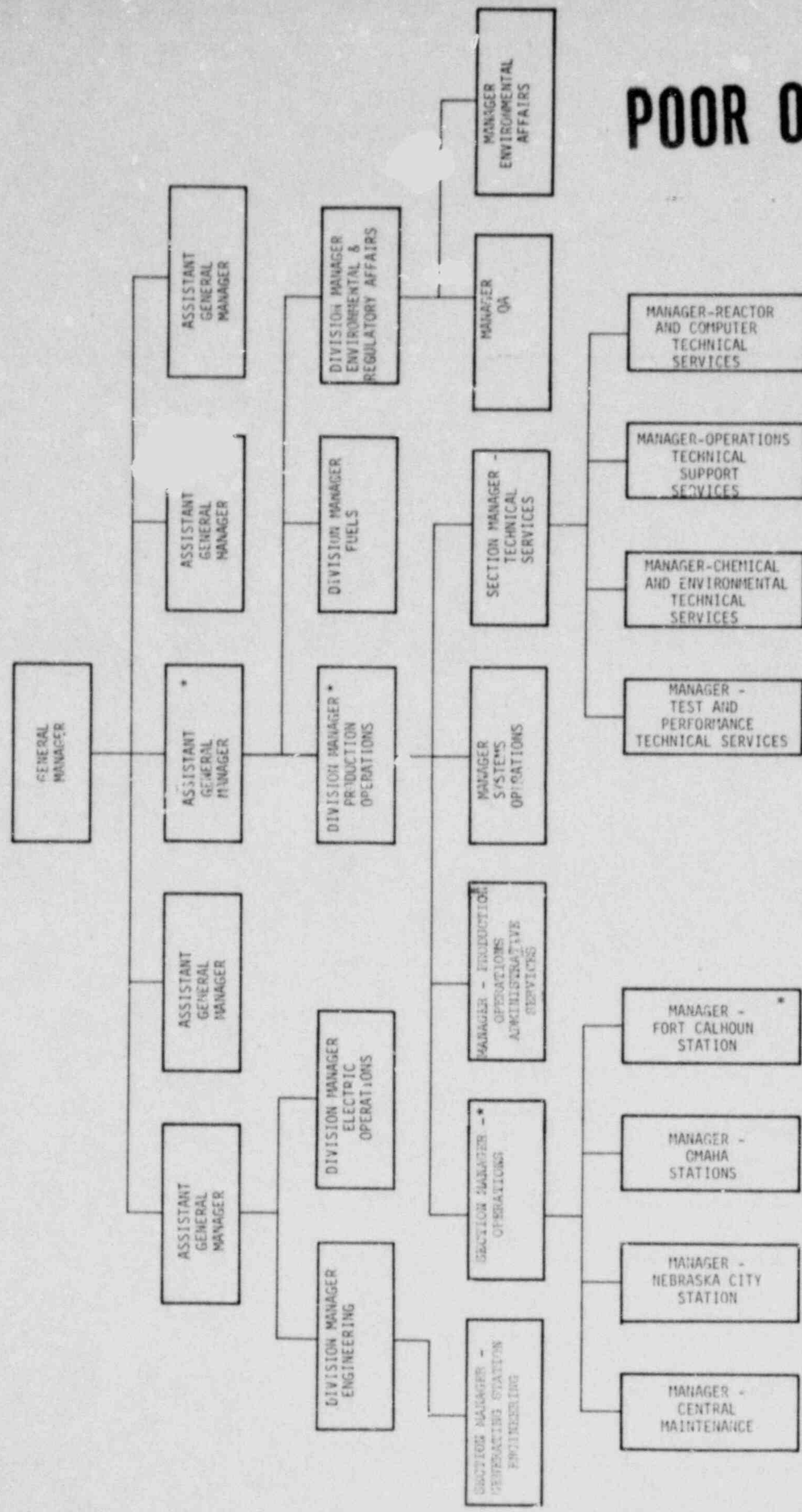
The reactor core shall approximate a right circular cylinder with an equivalent diameter of 106.5 inches and an active height of 128 inches. The reactor core shall normally consist of Zircaloy-4 clad fuel rods containing slightly enriched uranium in the form of sintered UO_2 pellets. The fuel rods shall 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 CEA's 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.

POOR ORIGINAL



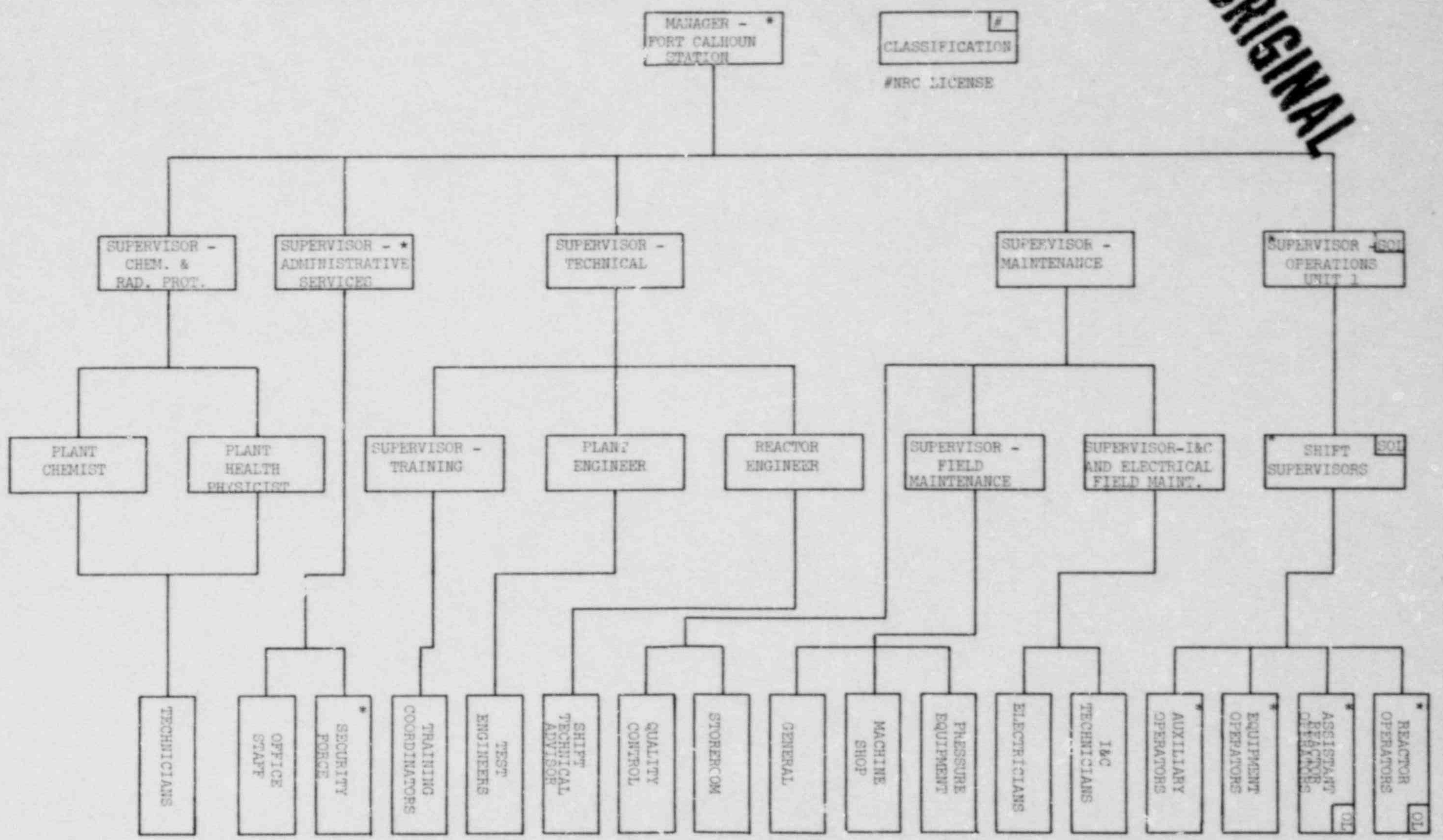
*Positions possessing fire protection responsibilities.

FORT CALHOUN TECHNICAL SPECIFICATIONS	OPPD SUPPORT STAFF	FIGURE 5-1

Amendment No. 1, 1, 35

Figure 5-1A

DELETED



*Positions Possessing Fire Protection Responsibilities

FORT CALHOUN TECHNICAL SPECIFICATIONS	FORT CALHOUN STATION ORGANIZATION CHART	FIGURE 5-2

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 5.1.1 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

5.5.2.2 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 advise 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.

5.5.2.8 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 5.5.2.8 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 an REPORTABLE OCCURRENCE:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 5.9.

(d) If $U_q > 7\%$ the total peaking uncertainty factor defined as $(1 + U_q)$ shall be used in place of the measurement-calculation factor of 1.07 in Specification 2.10.4(1).

(e) The maximum local peak linear heat rate in the core, Φmax, shall be determined and the incore detector alarms shall be adjusted to no greater than the following:

$$\text{Alarm Setting} = \frac{C * \Phi\text{allowed}}{\Phi\text{max}}$$

where

C = The detector signal converted to flux units when the reactor is operating at steady-state.

Φallowed = Linear Heat Rate (kw/ft) allowed by Specification 2.10.4(1) and adjusted as required by Specification 6.3(1)(d).

Φmax = The maximum local peak linear heat rate (kw/ft) measured at the same reactor conditions as C above.

(2) If the incore detector system is not operable within the interval specified, the peak linear heat rate shall be monitored by excore detectors per Specification 2.10.4(1)(c) and the surveillance requirements of Specification 3.10(5) are deleted.

Basis

Operation of the incore detector system for peak linear heat rate monitoring and surveillance of F_R^T and F_{xy}^T with less than 75% of the strings operable requires additional measures to compensate for degradation of the incore instrument system. Periodic comparisons between calculated and measured power distributions are made to confirm the core is depleting as designed. The measurement uncertainties are computed to assure the assumptions made in the setpoint analysis are valid. The uncertainties are computed using the methods given in the reference.

If the determined uncertainties exceed the uncertainties used in the setpoint and safety analysis, the measured values of F_R and F_{xy} are augmented by the appropriate uncertainty. These new values of F_R and F_{xy} are then used to verify compliance with Specifications 2.10.4(2) and 2.10.4(3). This assures that the product of the radial peaking factors and their appropriate uncertainties are less than the values used in determining the setpoints.

DISCUSSION

The proposed license change is submitted to:

(1) Correct the Table of Contents (page ii).

(2) Correct a typographical error (page 2-9).

(3) Correct a table numbering problem (duplicate Tables 2-6).

(4) Provide clarification of bypass conditions for three Reactor Protective System (RPS) trip functions (Table 2-2).

(5) Delete turbine runback test reference (Table 3-1).

(6) Correct a typographical error (Table 3-5).

(7) Update reactor core and control design features to reflect changes in fuel design with Cycle 6 reload.

(8) Update Omaha Public Power District's organizational charts (Figures 5-1, 5-1A, and 5-2).

(9) Revise the Safety Audit and Review Committee membership consistent with the new organization (pages 5-5, 5-6, and 5-8).

A more detailed discussion of each item above is provided.

Table of Contents (page ii). The Table of Contents has been revised to add section 5.5.3. Specification 5.5.3 was added by Amendment No. 38.

Typographical Errors (page 2-9). Two typographical errors were noted on page 2-9. The first sentence should read: "The potential dose at the site boundary..." Also, item (1) title should be: "(1) Steam Generator Tube Rupture". These two items have been corrected.

Typographical Errors (page 2-34). Change AEC to NRC. Also, change Denver to Arlington, Texas.

Table Number Change. Two tables numbered 2-6 are presently in the Technical Specifications. The first is on page 2-57f and is referenced in Specification 2.10.4 (pages 2-57c and 2-57d). The second Table 2-6 is on pages 2-75 through 2-88 and is referenced in Specification 2.18 (pages 2-73 and 2-74). The table of page 2-57f has been renumbered to Table 2-9 and references to it corrected accordingly.

RPS Bypass Conditions. For each channel, a single bistable provides for automatic activation of the Axial Power Distribution (APD) and Loss of Load reactor trips and deactivation (bypass) of the high rate trip-wide range log channels as reactor power increases above 15% + %. Table 2-2 lists permissible bypass conditions as follows:

9. Axial Power Distribution: Below 15% of rated power.

10. High Rate Trip-Wide Range Long Channels: Below 10^{-4}% and above 15% of rated power.

11. Loss of Load: Below 15% of rated power.

Using the same bistable to provide the automatic bypass function, condition 10 cannot be met concurrently with items 9 and 11. In order to eliminate any confusion, note (e) has been added to Table 2-2.

This proposed change does not constitute an unreviewed safety question, as detailed in the attached evaluation (Form FC-154).

Table Corrections. Tables 2-7 and 2-8 have been corrected to agree with Fire Protection System since installation of fire protection modifications.

Turbine Runback Test (Table 3-1). Amendment No. 32 deleted the turbine runback requirement from the Fort Calhoun Station Unit No. 1 Technical Specifications. This editorial change updates the surveillance testing requirements.

Typographical Error (Table 3-5). Item 14, under the component column, lists 12 valves; the ninth in succession being HCV-372. This is a typographical error and has been revised to the correct valve number, HCV-312.

Reactor Core Design (Specification 4.3.2). The reactor core design has changed with the Cycle 6 reload. Specifically, there are no burnable poison rods and the fuel distribution has been changed such that there are now 23,408 fuel rods in the core and 120,610 pounds of UO_2.

Organizational Charts. The District's Staff Support organizational charts (Figures 5-1, 5-1A, and 5-2) have been revised to reflect the District's current organization. Figure 5-1A has been deleted.

Safety Audit and Review Committee (SARC) Revision (Specification 5.5.2). The proposed license change revises the SARC membership listing to reflect Omaha Public Power District's current organization and to specifically designate those consultants and staff members assigned to the SARC. This is strictly an administrative change and does not change the responsibilities or functions of the SARC. Also, 5.5.2.8.e. is revised to delete fire protection program, since this requirement is redundant to Technical Specification 5.5.3.a.

Interim Technical Specification 6.4. Typographical error in section 6.4(2). Change is made to reference correct Technical Specification section.

The eleven proposed license changes discussed above do not constitute safety or environmental issues and do not constitute an unreviewed safety question. Additionally, all proposed license changes are consistent with the requirements of Combustion Engineering Standard Technical Specifications.

JUSTIFICATION FOR FEE CLASSIFICATION

The proposed amendment is deemed to be a Class II Amendment, within the meaning of 10 CFR 170.22. The requested changes are strictly administrative in nature and represent no safety considerations.

ATTACHMENT C