DUKE POWER COMPANY
P.O. BOX 33189
CHARLOTTE, N.C. 28242

HAL B. TUCKER
VICE PRESIDENT
NUCLEAR PRODUCTION

TELEPHONE
(704) 373-4531

April 26, 1984

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Ms. E. G. Adensam, Chief
Licensing Branch No. 4

Re: Catawba Nuclear Station, Unit 1
Docket No. 50-413
Proof and Review Technical Specifications

Dear Mr. Denton:

Attachments 1-5 of this letter contain proposed amendments to the Proof and Review Technical Specifications for Catawba Unit 1. Each attachment contains the proposed changes and a discussion of the justification.

Very truly yours,

Hal B. Tucker

RWO/php

Attachments

cc: Mr. James P. O'Reilly, Regional Administrator
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, NW, Suite 2900
Atlanta, Georgia 30303

NRC Resident Inspector
Catawba Nuclear Station

Mr. Robert Guild, Esq.
Attorney-at-Law
P. O. Box 12097
Charleston, South Carolina 29205

Mr. Jesse L. Riley
Carolina Environmental Study Group
854 Henley Place
Charlotte, North Carolina 28207

Palmetto Alliance
2135½ Devine Street
Columbia, South Carolina 29205

B001
1/1

Attachment 1

Proposed Amendment to Catawba Unit 1 Proof and Review

Technical Specification 3.3.3.8 Concerning

Fire Detection Instrumentation

The proposed change would reduce the surveillance requirement for the fixed temperature/rate of rise fire detection instruments.

The current Catawba Technical Specifications would require testing each accessible detector every 6 months. The test for a nonrestorable (fixed temperature) heat detector is destructive. The fixed temperature and the rate of rise (restorable) detectors are in the same enclosure so that when a fixed temperature detector is tested (and destroyed) than the rate of rise portion becomes useless.

This Technical Specification has been accepted for the McGuire Units 1 and 2 Technical Specifications.

PROOF & REVIEW COPY

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 As a minimum, the fire detection instrumentation for each fire detection zone shown in Table 3.3-11 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the fire detection instrument is required to be OPERABLE.

ACTION:

a. With any, but not more than one-half the total in any fire zone, Function A fire detection instruments shown in Table 3.3-11 inoperable, restore the inoperable instrument(s) to OPERABLE status within 14 days or within ~~the next~~ 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5.

b. With more than one-half of the Function A fire detection instruments in any fire zone shown in Table 3.3-11 inoperable, or with any Function B fire detection instruments shown in Table 3.3-11 inoperable, or with any two or more adjacent fire detection instruments shown in Table 3.3-11 inoperable, within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.8.1 Each of the above required fire detection instruments which are accessible during plant operation shall be demonstrated OPERABLE at least once per 6 months by performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST. Fire detectors which are not accessible during plant operation shall be demonstrated OPERABLE by the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

INSERT FROM NEXT PAGE

4.3.3.8.2 The NFPA Standard 72D supervised circuits supervision associated with the detector alarms of each of the above required fire detection instruments shall be demonstrated OPERABLE at least once per 6 months.

~~4.3.3.8.3 The nonsupervised circuits, associated with detector alarms, between the instrument and the control room shall be demonstrated OPERABLE at least once per 31 days.~~

DUKE POWER COMPANY

Form 00184 (6-81)

Dev./Station Unit File No.

Subject

By Date

Sheet No. of Problem No. Checked By Date

INSERT FOR 4.3.3.8.1

Each of the above required fixed temperature/rate of rise detection instruments shall be demonstrated OPERABLE as follows:

a. For nonrestorable spot-type detectors, at least two detectors out of every hundred, or fraction thereof, shall be removed every 5 years and functionally tested. For each failure that occurs on the detectors removed, two additional detectors shall be removed and tested.

b. For restorable spot-type heat detectors, at least one detector on each signal initiating circuit shall be demonstrated OPERABLE at least once per 6 months by performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST. Different detectors shall be selected for each test. Fire detectors which are not accessible during plant operation shall be demonstrated OPERABLE by the performance of a TRIP ACTUATING DEVICE OPERATIONAL TEST during each COLD SHUTDOWN exceeding 24 hours unless performed in the previous 6 months.

Attachment 2

Proposed Amendment to Catawba Unit 1 Proof and Review Technical Specification Table 3.6-2 Concerning the Containment Isolation Valves

The proposed changes would allow Main Steam Isolation Valves SM-70, SM-71, SM-72 and SM-73 to be opened on an intermittent basis under administrative control.

When shutting a unit down, the steam generators are placed in hot layup which begins with a S/G temperature in excess of 250°F. As the steam generator is cooled, the nitrogen inerting system is to be activated before the steam generator shell-side pressure has dropped below 25 psig in order that nitrogen gas will be admitted to maintain a positive pressure in the S/G shell at all times.

The capability exists to align nitrogen to the S/G through a 1" line into the main steam line. Containment integrity requirements dictate that this pathway cannot be aligned until the reactor coolant temperature drops below 200°F (Mode 5).

Since the unit is in Mode 4 and approaching Mode 5, and the nitrogen line is a relatively small line, we request this Technical Specification change in order that the nitrogen can be aligned via the present pathway before the reactor coolant temperature drops below 200°F.

PROOF & REVIEW COPY

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Sec.)
3. Manual		
NC-141	NC Pump H_2 Drain Tank Pump Discharge	N.A.
NC-142	NC Pump H_2 Drain Tank Pump Discharge	N.A.
NV-862#	Pzr. Aux. Spray Transient Line	N.A.
NI-3	Boron Injection Tank Line to Cold Legs	N.A.
FW-11	Refueling Water Pump Suction	N.A.
FW-13	Refueling Water Pump Suction	N.A.
CF-91#	Feedwater 1A	N.A.
CF-93#	Feedwater 1B	N.A.
CF-95#	Feedwater 1C	N.A.
CF-97#	Feedwater 1D	N.A.
CA-121#	Aux. Feedwater 1A	N.A.
BW-1#	Aux. Feedwater 1A	N.A.
CA-120#	Aux. Feedwater 1B	N.A.
BW-26#	Aux. Feedwater 1B	N.A.
CA-119#	Aux. Feedwater 1C	N.A.
BW-17#	Aux. Feedwater 1C	N.A.
CA-118#	Aux. Feedwater 1D	N.A.
BW-10#	Aux. Feedwater 1D	N.A.
SM-16#	Main Steam 1A	N.A.
SM-73#*	Main Steam 1A	N.A.
SM-105#	Main Steam 1A	N.A.
SM-121#	Main Steam 1A	N.A.
SM-143#	Main Steam 1A	N.A.
SM-72#*	Main Steam 1B	N.A.
SM-104#	Main Steam 1B	N.A.
SM-120#	Main Steam 1B	N.A.
SM-142#	Main Steam 1B	N.A.
SH-1#	Main Steam 1B	N.A.
SM-17#	Main Steam 1B	N.A.
SM-18#	Main Steam 1C	N.A.
SM-71#*	Main Steam 1C	N.A.

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (Sec.)
3. Manual (Continued)		
SM-103#	Main Steam 1C	N.A.
SM-119#	Main Steam 1C	N.A.
SM-141#	Main Steam 1C	N.A.
SA-4#	Main Steam 1C	N.A.
SM-19#	Main Steam 1D	N.A.
SM-70#*	Main Steam 1D	N.A.
SM-102#	Main Steam 1D	N.A.
SM-118#	Main Steam 1D	N.A.
SM-140#	Main Steam 1D	N.A.
WE-20*	Cont Bldg Supply Isol	N.A.
WE-22*	Cont Bldg Supply Isol	N.A.
FW-4*	Refueling Water	N.A.
NV-862#*	Pressurizer Auxiliary Spray ND Outside Containment	N.A.
WLA-21#*	Steam Generator Drain Pump Discharge Outside Containment Isolation	N.A.
WLA-24#*	Steam Generator Drain Pump Discharge Outside Containment Isolation	N.A.

TABLE NOTATIONS

* May be opened on an intermittent basis under administrative control.

** Valve also receives a High Radiation (H) and/or a High Relative Humidity isolation signal.

Not subject to Type C leakage tests.

NOTE: Times are for valve operation only, and do not include any sensor response or circuit delay times. See Specification 3/4 3.2 for system actuation response times.

Attachment 3

Proposed Amendment to Catawba Unit 1 Proof and Review

Technical Specification 3.11.1.5 Concerning the

Chemical Treatment Ponds

The proposed changes to the Chemical Treatment Pond Technical Specification are to take account of the addition of bead resin to the ponds. The current Technical Specification is based upon the addition of powdered resin only.

PROOF & REVIEW COPY

RADIOACTIVE EFFLUENTS

CHEMICAL TREATMENT PONDS

LIMITING CONDITION FOR OPERATION

3.11.1.5 The quantity of radioactive material contained in each chemical treatment pond shall be limited by the following expression:

$$\frac{264}{V} \cdot \sum_j \frac{A_j}{C_j} < 1.0$$

excluding tritium and dissolved or entrained noble gases,

Where:

A_j = pond inventory limit for single radionuclide "j", in Curies;

C_j = 10 CFR Part 20, Appendix B, Table II, Column 2, concentration for single radionuclide "j", microCuries/ml;

V = design volume of liquid and slurry in the pond, in gallons; and

264 = conversion unit, microCuries/Curie per milliliter/gallon.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above listed ponds exceeding the above limit, immediately suspend all additions of radioactive material to the pond and initiate corrective action to reduce the contents to within the limit.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.5 The quantity of radioactive material contained in each batch of resin/water slurry ~~(used powdex resin)~~ to be transferred to the chemical treatment ponds shall be determined to be within the above limit by analyzing a representative sample of the batch ~~slurry~~, and batches to be transferred to the chemical treatment ponds shall be limited by the expression:

$$\sum_j \frac{\cancel{Q_j}\ c_j}{C_j} < \cancel{0.6\ \frac{\mu Ci/gm}{\mu Ci/ml}}\ 0.006$$

Where:

~~Q_j = concentration of radioactive materials in wet, drained slurry (used powdex resin) for radionuclide "j", excluding tritium, dissolved or entrained noble gases, and radionuclides with less than an 8-day half-life. The analysis shall include at least Ce-144, Cs-134, Cs-137, Co-58 and Co-60, in microCuries/gram. Estimates of the Sr-89 and Sr-90 batch concentration shall be included based on the most recent monthly composite analysis (within 3 months); and~~

C_j = 10 CFR Part 20, Appendix B, Table II, Column 2, concentration for single radionuclide "j", in microCuries/milliliter.

c_j = radioactive resin/water slurry concentration for radionuclide "j" entering the UNRESTRICTED AREA chemical treatment ponds, in microCuries/milliliter; and

PROOF & REVIEW COPY

RADIOACTIVE EFFLUENTS

BASES

3/4.11.1.4 LIQUID HOLDUP TANKS

The tanks included in this specification are all those outdoor radwaste tanks that are not surrounded by liners, dikes or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System.

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an UNRESTRICTED AREA.

3/4.11.1.5 CHEMICAL TREATMENT PONDS

The inventory limits of the chemical treatment ponds (CTP) are based on limiting the consequences of an uncontrolled release of the pond inventory. The expression in Specification 3.11.1.5 assumes the pond inventory is uniformly mixed, that the pond is located in an uncontrolled area as defined in 10 CFR Part 20, and that the concentration limit in Note 1 to Appendix B of 10 CFR Part 20 applies.

The batch limits of the resin/water slurry transferred to the chemical treatment ponds assure that radioactive material ~~in the slurry~~ transferred to the CTP ~~are~~ is "as low is reasonably achievable" in accordance with 10 CFR 50.36a. The expression in Specification 4.11.1.5 assures no batch ~~of slurry~~ will be transferred to the CTP unless the sum of the ratios of the activity of the radionuclides to their respective concentration limitation is less than the ratio of the 10 CFR Part 50, Appendix I, Section II.A, total body dose level to the 10 CFR 20.105(a), whole body dose limitation, or that:

$$\sum_j \frac{c_j}{C_j} < \frac{3\ \text{mrem/yr}}{500\ \text{mrem/yr}} = 0.006$$

Where:

c_j = radioactive resin/water slurry concentration for radionuclide "j" entering the ~~unrestricted area~~ CTP, in microCuries/milliliter; and

INSERT FROM NEXT PAGE

C_j = 10 CFR Part 20, Appendix B, Table II, Column 2, concentration for single radionuclide "j", in microCuries/milliliter.

~~For the design of filter/demineralizers using powder resin, the slurry wash volume and the weight of resin used per batch is fixed by the cell surface area, and the slurry volume to resin weight ratio is constant at 100 milliliters/gram of wet, drained resin with a moisture content of approximately 55 to 60% (bulk density of about 58 pounds per cubic feet). Therefore,~~

DUKE POWER COMPANY

Form 00184 (6-81)

Dev./Station ______ Unit ____ File No. ______

Subject ______

______ By ______ Date ______

Sheet No. ___ of ___ Problem No. ______ Checked By ______ Date ______

INSERT FOR BASES 3/4.11.1.5

The filter/demineralizers using powdered resin and the blowdown demineralizer are backwashed or sluiced to a holding tank. The tank will be agitated and recirculated to obtain a representative sample of the resin inventory in the tank. A known weight of the wet, drained resin (moisture content approximately 55 to 60%, bulk density of about 58 pounds per cubic foot) will then be counted. The concentration of the resin slurry to be pumped to the chemical treatment ponds will then be determined by the formula:

$$c_j = \frac{Q_j W_R}{V_T}$$

Where:

Q_j = concentration of radioactive materials in wet, drained resin for radionuclide "j", excluding tritium, dissolved or entrained noble gases, and radionuclides with less than an 8-day half-life. The analysis shall include at least Ce-144, Cs-134, Cs-137, Co-58 and Co-60, in microCuries/gram. Estimates of the Sr-89 and Sr-90 batch concentration shall be included based on the most recent monthly composite analysis (within 3 months);

W_R = total weight of resin in the storage tank in grams (determined from chemistry logs procedures); and

V_T = total volume of resin water mixture in storage tank to be transferred to the chemical treatment poinds in milliliters.

PROOF & REVIEW COPY

RADIOACTIVE EFFLUENTS

BASES

CHEMICAL TREATMENT PONDS (Continued)

$$\cancel{\sum_j \frac{C_j}{\bar{C}_j} = \sum_j \frac{Q_j}{C_j\,(10^2\ \mathrm{ml/gm})} < 0.006,\ \text{and}}$$

$$\cancel{\sum_j \frac{C_j}{\bar{C}_j} < 0.6\ \frac{\mu\mathrm{ci/gm}}{\mu\mathrm{ci/ml}}}$$

~~Where the terms are defined in Specification 4.11.1.5.~~

The batch limits provide assurance that activity input to the CTP will be minimized, and a means of identifying radioactive material in the inventory limitation of Specification 3.11.1.5.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

This specification is provided to ensure that the dose at any time at and beyond the SITE BOUNDARY from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 to UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA, either within or outside the SITE BOUNDARY, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR 20.106(b)). For MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of that MEMBER OF THE PUBLIC will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. Examples of calculations for such MEMBERS OF THE PUBLIC, with the appropriate occupancy factors, shall be given in the ODCM. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY to less than or equal to 500 mrems/year to the whole body or to less than or equal to 3000 mrems/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems/year.

This specification applies to the release of radioactive materials in gaseous effluents from all units at the site.

The required detection capabilities for radioactive material in gaseous waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L.A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J.K., "Detection

Attachment 4

Proposed Amendment to Catawba Unit 1 Proof and Review

Technical Specification 3.11.2.6 Concerning the

Gas Storage Tanks

The proposed change would raise the limit on the amount of radioactivity contained in each gas storage tank from 49,000 curies to 97,000 curies.

This proposed limit was developed utilizing the guidance contained in Standard Review Plan 11.3, Branch Technical Position ETSB 11-5. The previous limit was developed using the guidance contained in Standard Review Plan 15.7.1 which has been deleted.

PROOF & REVIEW COPY

RADIOACTIVE EFFLUENTS

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to ~~40,000~~ 97,000 Curies of noble gases (considered as Xe-133 equivalent).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.7.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank.

Attachment 5

Proposed Amendment to Catawba Unit 1 Proof and Review

Technical Specifications

Supplying Missing Information and Correcting Typographical Errors

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	4.56	0	≤109% of RTP*	≤111.2% of RTP*
b. Low Setpoint	8.3	4.56	0	≤25% of RTP*	≤27.2% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	1.6	0.5	0	≤5% of RTP* with a time constant ≥ 2 seconds	≤6.3% of RTP* with a time constant ≥ 2 seconds
4. Power Range, Neutron Flux, High Negative Rate	1.6	0.5	0	≤5% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.4	0	≤25% of RTP*	≤31% of RTP*
6. Source Range, Neutron Flux	17.0	10	0	≤10^5 cps	≤1.4×10^5 cps
7. Overtemperature ΔT	6.4	3.92	2.2	See Note 1	See Note 2
8. Overpower ΔT	4.6	1.4	1.2	See Note 3	See Note 4
9. Pressurizer Pressure-Low***	3.0	0.71	1.5	≥1945 psig	≥1934 psig
10. Pressurizer Pressure-High	3.1	0.71	1.5	≤2385 psig	≤2396 psig
11. Pressurizer Water Level-High	5.0	2.18	1.5	≤92% of instrument span	≤93.8% of instrument span
12. Reactor Coolant Flow-Low	2.5	1.77	0.6	≥90% of loop design flow**	≥88.8% of loop design flow**

*RTP = RATED THERMAL POWER

**Loop design flow = 96,900 gpm

*** Includes a lead time constant of 2 seconds and a lag time constant of 1 second. Channel calibration shall ensure that these time constants are adjusted to these values.

Attachment 5 Page 1

PROOF & REVIEW COPY

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
8. Auxiliary Feedwater (Continued)					
c. Steam Generator Water Level - Low-Low	12	12.18	1.5	17 >~~12%~~ of span from 0% to 30% RTP increasing linearly to >54.9% of span from 30% to 100% RTP	>~~10.25%~~ of span from 0% to 30% RTP increasing linearly to >53.15% of span from 30% to 100% RTP
d. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.				
e. Loss-of-Offsite Power	N.A.	N.A.	N.A.	> 3500 ~~(4000)~~ V	> 3200 ~~(3092)~~ V
f. Trip of All Main Feedwater Pumps	N.A.	N.A.	N.A.	N.A.	N.A.
g. Auxiliary Feedwater Suction Pressure-Low	N.A.	N.A.	N.A.	> 2 psig	> 1 psig
9. Containment Sump Recirculation					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. Refueling Water Storage Tank Level-Low Coincident With Safety Injection	N.A.	N.A.	N.A.	> 120 inches	> 114 inches
	See Item 1. above for all Safety Injection Setpoints and Allowable Values.				

Attachment 5 Page 2

PROOF & REVIEW COPY

THIS PAGE OPEN PENDING RECEIPT OF INFORMATION FROM THE APPLICANT

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
10. Loss of Power					
a. 4 kV Bus Undervoltage- ~~Grid Degraded Voltage~~ Loss of Voltage	N.A.	N.A.	N.A.	≥ 3500 ~~± 175~~ volts ~~with a 0.5 ± 0.5 second time delay~~	≥ 3200 volts
b. 4 kV Bus Undervoltage — Grid Degraded Voltage	N.A.	N.A.	N.A.	≥ 3744 volts	≥ 3669 volts
11. Control Room Area Ventilation Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Loss-of-Offsite Power	N.A.	N.A.	N.A.	N.A.	N.A.
12. Containment Air Return and Hydrogen Skimmer Operation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure-High-High	3.0	0.71	1.5	≤3 psig	<3.17 psig

PROOF & REVIEW COPY

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
13. Annulus Ventilation Operation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.				
14. Nuclear Service Water Operation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Spray	See Item 2. above for all Containment Spray Setpoints and Allowable Values.				
d. Phase "B" Isolation	See Item 3.b. above for all Phase "B" Isolation Setpoints and Allowable Values.				
e. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.				
15. Emergency Diesel Generator Operation (Diesel Building Ventilation Isolation)					
a. Manual Initiation	N.A	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Loss-of-Offsite Power	N.A.	N.A.	N.A.	3500 ≥~~(4800)~~V	3200 ≥~~[illegible]~~V
d. Safety Injection	See Item. 1 above for all Safety Injection Setpoints and Allowable Values.				

THIS PAGE OPEN PENDING RECEIPT OF INFORMATION FROM THE APPLICANT

PROOF & REVIEW COPY

Attachment 5 Page 4

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
16. Auxiliary Building Ventilation Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Loss-of-Offsite Power	N.A.	N.A.	N.A.	3500 ≥ ~~(4000)~~V	3200 ≥ ~~(4692)~~V
d. Safety Injection	See Item 1. above for all Safety Injection Setpoints and Allowable Values.				
17. Diesel Building Ventilation Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Emergency Diesel Generator Operation	See Item 15. above for all Emergency Diesel Generator Operation Setpoints and Allowable Values.				
18. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	≤ 1955 psig	≤ 1944 psig
b. Low-Low T_{avg}, P-12	N.A.	N.A.	N.A.	N.A.	N.A.
c. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	N.A.
d. Steam Generator Level, P-14	See Item 5. above for all Steam Generator Water Level Trip Setpoints and Allowable Values.				

THIS PAGE [illegible] RECEIPT OF
INFORMATION [illegible]

PROOF & REVIEW COPY

PROOF & REVIEW COPY

TABLE 4.4-4 (Continued)

TABLE NOTATIONS

#Until the specific activity of the Reactor Coolant System is restored within its limits.

*Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.

**A gross radioactivity analysis shall consist of the quantitative measurement of the total specific activity of the reactor coolant except for radionuclides with half-lives less than 10 minutes and all radioiodines. The total specific activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities in the sample within 2 hours after the sample is taken and extrapolated back to when the sample was taken. Determination of the contributors to the gross specific activity shall be based upon those energy peaks identifiable with a 95% confidence level. The latest available isotopic decay data may be used for pure beta-emitting radionuclides.

***A radiochemical analysis for $\bar{E}$ shall consist of the quantitative measurement of the specific activity for each radionuclide, except for radionuclides with half-lives less than 10 minutes and all radioiodines, which is identified in the reactor coolant. The specific activities for these individual radionuclides shall be used in the determination of $\bar{E}$ for the reactor coolant sample. Determination of the contributors to $\bar{E}$ shall be based upon those energy peaks identifiable with a 95% confidence level.