DUKE POWER COMPANY
POWER BUILDING
422 SOUTH CHURCH STREET, CHARLOTTE, N. C. 28242

WILLIAM O. PARKER, JR.
VICE PRESIDENT
STEAM PRODUCTION

TELEPHONE: AREA 704
373-4083

July 15, 1982

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
Docket Nos. 50-413 and 50-414

Dear Mr. Denton:

In order to facilitate the completion of the review of the Catawba FSAR, Duke Power Company is transmitting herewith responses, revised responses, or partial responses to the following FSAR questions:

123.1	220.35	810.2	810.33	ICSB 29
123.2	241.10	810.5	810.34	ICSB 30
123.3	410.10	810.17	810.35	ICSB 54
123.4	410.23	810.18	810.36	ICSB 70
123.5	410.25	810.19	810.39	ICSB 98
123.6	440.T.8	810.22	ICSB 6	
123.7	730 A-17	810.26	ICSB 17	

These responses will be included in FSAR Revision 6.

Very truly yours,

William O. Parker, Jr.

B001
Aper. D.st.
SEND Drwgs to:
PM

ROS/php
Attachment

cc: Mr. James P. O'Reilly, Regional Administrator
U. S. Nuclear Regulatory Commission
Region II
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Mr. P. K. Van Doorn
NRC Resident Inspector
Catawba Nuclear Station

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Mr. Henry A. Presler, Chairman
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DESIGN COMPARISON

Catawba Nuclear Station - Comparison with McGuire Nuclear Station and Watts Bar Nuclear Power Plant

CHAPTER NUMBER	CHAPTER TITLE SYSTEM/COMPONENT	REFERENCES (FSAR)	SIGNIFICANT SIMILARITIES	SIGNIFICANT DIFFERENCES
6.0	Engineered Safety Features			
	Containment	Sections 3.8.2 and 6.2.1	McGuire, Watts Bar	None
	Containment Spray System	Section 6.2.2	McGuire, Watts Bar	None
	Hydrogen Purge System	Section 6.2.5	McGuire, Watts Bar	None
	Hydrogen Recombiners	Section 6.2.5	McGuire, Watts Bar	None
	Emergency Core Cooling System	Section 6.3	McGuire, Watts Bar except as noted	The following has been added or changed: · New ANS Safety Classification.
	Ice Condenser	Section 6.7	McGuire, Watts Bar	None
7.0	Instrumentation and Controls			
	Reactor Trip System	Section 7.2	System functions are similar to McGuire, Watts Bar	None
ICSB Q70	Engineered Safety Features System	Section 7.3	Systems functions are similar to McGuire	Watts Bar is similar except steamline break protection is S.I. on high steam line differential pressure or high steam flow coincident with low steam line pressure or lo lo T_{avg}.

Response to TMI Concerns

The Catawba Work Request Program governs all maintenance activities performed at Catawba. These work requests describe the maintenance to be performed and the procedures for performing it. Upon completion of the maintenance all work requests are entered into the corporate computer. This program provides for portable historical records of all maintenance performed on safety-related systems.

C.1.17

The design of Catawba Nuclear Station does not feature safety injection initiation on coincident pressurizer level and pressure signals. Safety injection is initiated whenever the low pressurizer pressure trip setpoint is reached independent of pressurizer level (See Section 7.3).

II.K.2 COMMISSION ORDERS ON B&W PLANTS

II.K.2.13 THERMAL MECHANICAL REPORT - EFFECT OF HIGH-PRESSURE INJECTION ON VESSEL INTEGRITY FOR SMALL-BREAK LOSS-OF-COOLANT ACCIDENT WITH NO AUXILIARY FEEDWATER

Q440.T.8 WCAP-10019 which addresses the NRC requirements of detailed analysis of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater was submitted to the NRC on December 30, 1981 (OG-66). This WCAP was developed under the sponsorship of the Westinghouse Owners Group (WOG). On March 23, 1982 WOG letter OG-68 was submitted to the NRC which described the additional effort underway to resolve NRC comments and questions concerning WCAP-10019. Results of the program to date show that operating plants can withstand the limiting transients for the expected life of their vessels. Since WCAP-10019 only addresses operating plants, an additional effort is underway to address NTOL plants. This report should be available in early 1983.

II.K.2.17 POTENTIAL FOR VOIDING IN THE REACTOR COOLANT SYSTEM DURING TRANSIENTS

Westinghouse (in support of the Westinghouse Owners Group) has performed a study which addresses the potential for void formation in Westinghouse designed nuclear steam supply systems during natural circulation cooldown/depressurization transients. This study has been submitted to the NRC by the Westinghouse Owners Group (Letter OG-57, dated April 20, 1981, R. W. Jurgensen (Chairman, Westinghouse Owners Group) to P.S. Check (NRC)) and is applicable to Catawba Nuclear Station.

Q440.T.2

In addition, the Westinghouse Owners Group has developed a natural circulation cooldown guideline that takes the results of the study into account so as to preclude void formation in the upper head region during natural circulation cooldown/depressurization transients, and specifies those conditions under which upper head voiding may occur. These Westinghouse Owners Group generic guidelines have been submitted to the NRC (Letter OG-64, dated November 30, 1981, R. W. Jurgensen (Chairman, Westinghouse Owners Group) to D. G. Eisenhut (NRC)). The generic guidance developed by the Westinghouse Owners Group (augmented as appropriate with plant specific considerations) will be utilized in the implementation of Catawba plant specific operating procedures.

For each wave type, the full ground motion was assumed for calculating pipe stresses. This is conservative, since the ground motion is actually the result of summing the contribution from each of the wave types.

B. Seismic shear wave propagation velocity - 2000 fps; compression wave propatation velocity - 4200 fps.

C. Maximum ground acceleration - Bedrock, 0.15g.
Backfill 0.40g.

D. Coefficient of subgrade reaction (K) for site backfill material (varies with pipe size).
Average value - 1,580 psi/in

Q220.35

E. Friction coefficient - 0.3

Q241.10 F. Poisson's ratio - 0.35 for soil, 0.25 for rock

G. Shear wave velocity of rock 7000 fps, Rayleigh wave velocity 6000 fps.

Information on the other properties of the subsurface materials is presented in Chapter 2.

The coefficient of subgrade reaction (K) was determined by the method described in Reference 16 and the seismic wave material velocity obtained from the site. Varying the shear wave velocity of the soil by ± 25% will result in an increase and decrease of the coefficient; this will respectively increase and decrease stress on the pipe. The resulting maximum stress value thus calculated is still less than the allowable pipe stress.

Pipe settlement is considered negligible. For further information see Section 2.5.4.10.3.

(b) Static effects of displacements among structures to which the piping is attached. The Nuclear Service Water piping penetrates structures supported on continuous rock -

Auxiliary Building
Diesel Generator Buildings
NSW/SNSW Pump Structure

and structures supported on partially weathered rock -

NSW Intake Structure (lake)
SNSW Intake Structure (pond)
SNSW Discharge Structures.

There is also a non-seismic connection to the Low Pressure Service Water System piping for discharge during normal operation. The appropriate differential movements of the structure during the earthquake are imposed on the piping, assuming a fixed end connection at the point of entry into the structure.

Table 5.3.1-6
Catawba Unit 1 Closure Head Bolting Material Properties

Closure Head Studs

Heat No.	Mat'l Spec. No.	Bar No.*	0.2 YS KSi	UNTS KSi	ELONG %	RA %	BHN	Energy At 10°F Ft-Lbs	Lateral Expansion Mils
35674	A540,B24	25K	142.5	160.4	18.8	57	341	42,40,40	12,16,28
35674	A540,B24	25T	136.9	157.1	18.0	57	331	38,38,39	12,8,12
35674	A540,B24	26K	143.9	161.8	19.0	57	331	46,46,46	32,28,28
35674	A540,B24	26T	138.2	162.7	19.4	56	341	44,44,43	32,24,28
35674	A540,B24	27K	141.5	161.8	18.2	56	331	40,39,39	12,28,12
35674	A540,B24	27T	141.5	160.5	18.4	56	331	44,43,44	16,24,24
35666	A540,B24	28K	143.6	162.8	18.4	52	321	48,48,46.5	28,24,24
35666	A540,B24	28T	145.0	164.9	18.0	52	331	45.5,46.5,49.5	20,20,24
35666	A540,B24	29K	145.7	163.8	18.2	55	341	42,40.5,40.5	24,16,20
35666	A540,B24	29T	145.7	163.8	18.4	55	331	38,38,40.5	20,16,24
Q123.3 35847	A540,B24	297K	143.6	160.4	18.0	59	321	49,52,52.5	32,24,24
35847	A540,B24	297K	145.7	161.8	18.0	59	341	49.5,53.5,53.5	32,32,35

Closure Head Nuts

Heat No.	Mat'l Spec. No.	Bar No.*	0.2 YS KSi	UNTS KSi	ELONG %	RA %	BHN	Energy At 10°F Ft-Lbs	Lateral Expansion Mils
36627	A540,B24	328K	133.5	153.7	20.0	61	331	60,62,58	35,45,47
36627	A540,B24	328T	135.9	154.7	20.0	60	331	60.5,67,62	39,47,43

Closure Head Washers

Heat No.	Mat'l Spec. No.	Bar No.*	0.2 YS KSi	UNTS KSi	ELONG %	RA %	BHN	Energy At 10°F Ft-Lbs	Lateral Expansion Mils
36512	A540,B24	206K	131.2	153.7	19.6	60	331	48.5,49.5,49	35,35,32
36512	A540,B24	306T	132.5	153.7	20.2	60	321	52,52.5,50.5	43,43,43

*K & T denote top and bottom of bar respectively.

Q123.4

Q123.6

TABLE 5.3.3-4

CATAWBA UNIT 1 REACTOR VESSEL BELTLINE REGION TOUGHNESS PROPERTIES

INTERMEDIATE SHELL
HEAT NO. 411343 (TANG)

TEMP. (°F)	ENERGY (ft/lb)	LAT. EXP. (MILS)	SHEAR (%)
-148	2.0	6	0
-148	3.0	2	0
-148	4.5	4	0
- 76	8.5	8	0
- 76	3.5	4	0
- 76	8.5	4	0
- 4	45.5	39	27
- 4	16.5	20	16
- 4	42.0	35	17
40	101.5	79	52
40	102.5	79	49
40	100.0	71	49
40	87.0	71	47
40	86.5	71	38
40	57.5	51	38
60	118.0	75	65
60	122.5	79	57
60	92.0	63	52
113	140.0	87	85
113	129.5	91	76
113	155.0	99	90
176	152.5	87	100
176	156.5	87	100
176	152.5	94	100

T_{NDT} = -40°F

RT_{NDT} = -8°F

LOWER SHELL
HEAT NO. 527708 (TANG)

TEMP. (°F)	ENERGY (ft/lb)	LAT. EXP. (MILS)	SHEAR (%)
-148	2.5	2	0
-148	3.5	4	0
-148	4.5	4	0
- 76	11.5	12	4
- 76	11.5	8	3
- 76	11.5	8	3
- 4	57.5	47	29
- 4	57.5	51	23
- 4	53.5	47	23
40	101.5	79	67
40	126.5	94	80
40	109.5	83	68
40	149.0	94	80
40	156.5	91	75
40	113.0	83	65
60	125.5	83	70
60	117.0	71	62
60	106.5	71	55
113	149.5	94	100
113	135.5	91	85
113	149.0	94	100
176	155.0	99	98
176	158.5	91	100
176	151.0	91	100

T_{NDT} = -13°F

RT_{NDT} = -13°F

INTER. TO LOWER SHELL WELD
WELD CODE NO. R747

TEMP. (°F)	ENERGY (ft/lb)	LAT. EXP. (MILS)	SHEAR (%)
-148	3.5	4	0
-112	8.5	12	16
- 76	21.5	23	33
- 40	45.0	39	43
- 22	57.5	47	55
- 4	54.5	43	47
32	92.5	71	76
68	104.5	81	92
86	113.0	91	89
122	123.5	83	93
140	144.0	99	100
158	129.0	94	98
176	130.0	87	98
212	126.5	87	100

T_{NDT} = 76°F
RT_{NDT} = 51°F

INTER TO LOWER SHELL WELD ROOT
WELD CORE NO. P710

10	57.5	43	47
10	43.0	39	47
10	39.0	55	55

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New Page

Q123.6

TABLE 5.3.3-5 (Page 1)

CATAWBA UNIT 1 REACTOR VESSEL BELTLINE REGION TOUGHNESS PROPERTIES

INTERMEDIATE SHELL COURSE

PLATE B8605-1

TEMP. (°F)	ENERGY (ft/lb)	LAT. EXP. (MILS)	SHEAR (%)
- 40	8	6	0
- 40	9	8	0
- 40	7	7	0
10	29	21	20
10	19	16	15
10	23	20	15
40	33	25	20
40	50	36	25
40	45	33	25
75	48	36	30
75	50	40	40
75	56	43	40
75	51	39	40
75	51	38	40
100	63	50	50
100	74	55	60
100	58	44	50
160	87	60	95
160	91	67	100
160	88	62	95

T_{NDT} = -10°F

RT_{NDT} = 15°F

PLATE B8605-2

TEMP. (°F)	ENERGY (ft/lb)	LAT. EXP. (MILS)	SHEAR (%)
- 40	5	4	0
- 40	4	4	0
- 40	6	5	0
10	15	12	5
10	22	18	10
10	14	12	5
40	25	22	15
40	37	27	20
40	37	26	20
75	42	31	25
75	50	37	30
75	37	30	25
100	70	54	50
100	58	42	40
100	63	46	40
160	78	60	95
160	83	63	95
160	86	64	100

T_{NDT} = -20°F

RT_{NDT} = 33°F

PLATE B8616-1

TEMP. (°F)	ENERGY (ft/lb)	LAT. EXP. (MILS)	SHEAR (%)
- 40	7	6	0
- 40	11	8	0
- 40	10	7	0
10	26	18	20
10	30	19	20
10	29	21	20
40	35	30	20
40	42	32	25
40	50	36	30
74	53	41	40
74	52	39	40
74	56	43	40
100	65	49	60
100	64	47	60
100	69	51	60
160	89	68	100
160	93	69	100
160	95	71	100

T_{NDT} = 0°F

RT_{NDT} = 12°F

TABLE 5.3.3-5 (Page 2)

CATAWBA UNIT 1 REACTOR VESSEL BELTLINE REGION TOUGHNESS PROPERTIES

INTERMEDIATE SHELL COURSE

PLATE B8806-1

TEMP. (°F)	ENERGY (ft/lb)	LAT. EXP. (MILS)	SHEAR (%)
- 40	16	11	0
- 40	15	9	0
- 40	13	10	0
10	36	26	15
10	40	28	20
10	38	28	20
40	48	37	30
40	49	36	30
40	46	34	30
74	60	45	50
74	53	41	50
74	70	52	70
100	77	62	90
100	72	59	80
100	84	64	100
160	82	61	100
160	85	65	100
160	82	64	100

T_{NDT} = -60°F

RT_{NDT} = 6°F

PLATE B8806-2

TEMP. (°F)	ENERGY (ft/lb)	LAT. EXP. (MILS)	SHEAR (%)
- 40	16	12	0
- 40	16	12	0
- 40	15	10	0
10	38	25	15
10	35	23	15
10	34	21	15
40	53	34	25
40	61	42	40
40	49	33	20
74	61	40	40
74	59	40	40
74	64	44	40
100	86	61	70
100	75	52	60
100	69	46	60
160	108	74	100
160	98	69	100
160	100	72	100

T_{NDT} = -40°F

RT_{NDT} = -10°F

PLATE B8806-1

TEMP. (°F)	ENERGY (ft/lb)	LAT. EXP. (MILS)	SHEAR (%)
- 40	14	9	0
- 40	9	6	0
- 40	18	13	5
10	35	23	15
10	26	19	10
10	44	33	20
40	39	24	15
40	43	31	20
40	44	31	20
74	53	39	30
74	58	43	35
74	72	50	60
100	75	50	70
100	81	52	70
100	88	59	80
160	102	69	100
160	103	71	100
160	110	73	100

T_{NDT} = -40°F

RT_{NDT} = 8°F

TABLE 5.3.3-5 (Page 3)

CATAWBA UNIT 2 REACTOR VESSEL BELTLINE REGION TOUGHNESS PROPERTIES

INTERMEDIATE AND LOWER SHELL VERTICAL WELD SEAMS AND GIRTH WELD SEAM

WELD CODE NO G1.45

Q123.6

TEMP (°F)	ENERGY (ft/lb)	LAT. EXP. (mils)
- 20	108	74
- 20	84	56
- 20	79	54
10	130	84
10	129	76
10	132	85

$T_{NDT} = -80°F$

$RT_{NDT} = -80°F$

In general, pressure-retaining equipment or piping is not permitted in the control complex enclosure. Several small hand-held fire extinguishers are located within the area for local fire control. Several self-contained type breathing apparatus are located within the habitability area. Areas of the Auxiliary Building and the Service Building which contain high-pressure equipment or piping have no direct interface with the control room enclosure.

6.4.2.5 Shielding Design

Refer to Section 12.3.2

6.4.3 SYSTEM OPERATIONAL PROCEDURES

The control complex is served by the Control Room Area Ventilation System. The operation of the ventilation system is the same for all plant operational modes. The pressurization system utilizes filters as described in Section 12.3.3 for filtration of pressurizing air during all modes of plant operation.

Each outside air intake for the pressurizing system is monitored for the presence of radioactivity, chlorine, and products of combustion during all plant operation modes. Isolation of each outside air intake occurs automatically upon indication of a high radiation level, or high concentration of chlorine or smoke in the intakes. In order to restore pressurization, station procedures instruct the operator to select the desired intake based on inspection of control room indicators.

6.4.4 DESIGN EVALUATION

6.4.4.1 Radiological Protection

Refer to Chapter 15.

6.4.4.2 Toxic Gas Protection

A hazards analysis for toxic material is presented in Section 2.2. The habitability of the control room was evaluated to determine if a site-related or offsite accident involving a release of hazardous chemicals exceeds the toxicity limits as specified in Regulatory Guide 1.78.

The survey has determined that the off-site storage and transportation of hazardous chemicals for industries in the vicinity of the plant is of sufficient distance (5 miles) from the plant that they do not present a hazard to the plant.

The chlorine detectors used in the control room outside air intakes to mitigate the consequences of toxic gas release, utilize Potassium Iodide electrolyte to sense concentrations of chlorine gas. Detector sensitivity is 1 ppm (by volume) in air. Response time is less than 5 seconds at a concentration of 5 ppm. Detector sensing points are located approximately 25 feet downstream of the control room outside air intakes. Isolation dampers are shown in Figure 9.4.1-1. Isolation damper closure time is given in Table 3.9.3-14. Outside air intake isolation is described in Section 9.4.1.1.

2. Core Thermal Overpower Trips

The specific trip functions generated are as follows:

a. Overtemperature ΔT trip

This trip protects the core against low DNBR and trips the reactor on coincidence as listed in Table 7.2.1-1 with one set of temperature measurements per loop. The setpoint for this trip is continuously calculated by analog circuitry for each loop by solving the following equation (for which the detailed time constants are given in Table 2.2-1 of the Technical Specifications):

ICSB Q17

$$\Delta T_{setpoint} = \Delta T_o \left[K_1 - K_2 \left(\frac{1 + \tau_1 s}{1 + \tau_2 s} \right) \left(T_{avg} - T^{o}_{avg} \right) + K_3 \left(P - 2235 \right) - f \left(\Delta\phi \right) \right]$$

Where:

ΔT_o = Indicated ΔT at rated thermal power

T_{avg} = Average reactor coolant temperature (°F)

T^{o}_{avg} = Indicated T_{avg} at rated thermal power

P = Pressurizer pressure (psig)

K_1 = Preset bias

K_2 = Preset gain which compensates for effects of temperature on the DNB limits

K_3 = Preset gain which compensates for the effect of pressure on the DNB limits

τ_1, τ_2 = Preset constants which compensate for piping and instrument time delay

s = Laplace transform operator (seconds $^{-1}$)

$f(\Delta\phi)$ = A function of the neutron flux difference between upper and lower long ion chambers. (Refer to Figure 7.2.1-2)

A separate long ion chamber unit supplies the flux signal for each overtemperature ΔT trip channel.

Increases in $\Delta\phi$ beyond a pre-defined deadband result in a decrease in trip setpoint. Refer to Figure 7.2.1-2.

The required one pressurizer pressure parameter per loop is obtained from separate sensors connected to three pressure taps at the top of the pressurizer. Four pressurizer pressure signals are obtained from the three taps by connecting one of the taps to two pressure transmitters. Refer to Section 7.2.2.3.3 for an analysis of this arrangement.

Figure 7.2.1-1, Sheet 5, shows the logic for overtemperature ΔT trip function.

b. Overpower ΔT trip

This trip protects against excessive power (fuel rod rating protection) and trips the reactor on coincidenece as listed in Table 7.2.1-1, with one set of temperature measurements per loop. The setpoint for each

ICSB Q17 channel is continuously calculated using the following equation (for which the detailed time constants are given in Table 2.2-1 of the Technical Specifications):

$$\Delta T_{setpoint} = \Delta T_o \left[K_4 - K_5 \left(\frac{\tau_3 s}{1 + \tau_3 s} \right) T_{avg} - K_6 \left(T_{avg} - T^{\circ}_{avg} \right) - f\left(\Delta\phi \right) \right]$$

Where:

ΔT_o = Indicated ΔT at rated thermal power

$f(\Delta\phi)$ = A function of the neutron flux difference between upper and lower long ion chamber section.

K_4 = A preset bias

K_5 = A constant which compensates for piping and instrument time delay

K_6 = A constant which compensates for the change in density flow and heat capacity of the water with temperature.

T°_{avg} = Indicated T_{avg} at rated thermal power

T_{avg} = Average reactor coolant temperature (°F)

τ_3 = Preset time constant (seconds)

s = Laplace transform operator (seconds^{-1})

The development of permissives P-6 and P-10 is shown on Figure 7.2.1-1, Sheet 4. All of the permissives ar digital; they are derived from analog signals in the nuclear power range and intermediate range channels.

See Table 7.2.1-2 for the list of protection system interlocks.

2. Blocks of Reactor Trips at Low Power

Interlock P-7 blocks a reactor trip at low power (below approximately 10 percent of full power) on a low reactor coolant flow in more than one loop, reactor coolant pump undervoltage, reactor coolant pump underfrequency, pressurizer low pressure or pressurizer high water level. See Figure 7.2.1-1, Sheets 5, 6, and 16, for permissive applications. The low power signal is derived from three out of four power range neutron flux signals below the setpoint in coincidence with two out of two turbine impulse changer pressure signals below the setpoint (low plant load). See Figure 7.2.1-1, Sheets 4 and 16, for the derivation of P-7.

The P-8 interlock blocks a reactor trip on low reactor coolant flow in any one loop when the plant is below approximately 50 percent of full power. The block action (absence of the P-8 interlock signal) occurs when three out of four neutron flux power range signals are below the setpoint. Thus, below the P-8 setpoint: the reactor is allowed to operate with one loop inactive and the low flow trip will not occur until two loops indicate low flow. See Figure 7.2.1-1, Sheet 4, for the derivation of P-8, and Sheets 5 and 16 for the applicable logic.

ICSB Q54 The P-9 interlock blocks a reactor trip following a turbine trip if the power is below 70%. See Figure 7.2.1-1, Sheet 16 for the functional implementation of the P-9 interlock. See Figure 7.2.1-1, Sheet 4 for the functional derivation of P-9.

See Table 7.2.1-2 for the list of protection system blocks.

7.2.1.1.4 Reactor Coolant Temperature Sensor Arrangement

The hot and cold leg resistance temperature detectors are inserted into reactor coolant bypass loops. A bypass loop from upstream of the steam generator to downstream of the steam generator is used for the hot leg resistance temperature detectors, and a bypass loop from downstream of the reactor coolant pump to upstream of the pump is used for the cold log resistance temperature detectors. The complete bypass loop is inside the containment. The resistance temperature detectors are located in manifolds and are directly inserted into the reactor coolant bypass loop flow without thermowells. Thermowells are not used in order to minimize the detector thermal lag. The bypass arrangement permits replacement of defective temperature elements while the plant is at hot shutdown without draining or depressurizing the reactor coolant loops.

Three sampling probes are installed in a cross sectional plane of each hot leg at approximately 120 degree intervals. Each of the sampling probes, which extends several inches into the hot leg coolant stream, contains five inlet orifices distributed along its length. In this way a total of fifteen locations in the hot leg stream are sampled providing a representative coolant temperature

the "CHANNEL TEST" annunciator in the control room. Bistable operation is tested by increasing the test signal to its trip setpoint and verifying bistable relay operation by control board annunciator and trip status lights.
ICSB Q30 The positive and negative rate trip bistables are tested using the same procedure.

It should be noted that a valid trip signal would cause the channel under test to trip at a lower actual reactor power level. A reactor trip would occur when a second bistable trips. No provision has been made in the channel test circuit for reducing the channel signal level below that signal being received from the Nuclear Instrumentation System detector.

A Nuclear Instrumentation System channel which can cause a reactor trip through one of the two protection logic (source or intermediate range) is provided with a bypass function which prevents the initiation of a reactor trip from that particular channel during the short period that it is undergoing test. These bypasses are annunciated in the control room.

The following periodic tests of the Nuclear Instrumentation System are performed:

a. Testing at plant shutdown
   1) Source range testing
   2) Intermediate range testing
   3) Power range testing

b. Testing between P-6 and P-10 permissive power levels
   1) Source range testing
   2) Power range testing

c. Testing above P-10 permissive power level
   1) Power range testing

Any deviations noted during the performance of these tests are investigated and corrected in accordance with the established calibration and trouble shooting procedures provided in the plant technical manual for the Nuclear Instrumentation System. Control and protection trip settings are indicated in the plant technical manual under precautions, limitations and setpoints.

For additional background information on the Nuclear Instrumentation System see Reference 2.

Solid State Logic Testing

The reactor logic trains of the Reactor Trip System are designed to be capable of complete testing at power. After the individual channel analog testing is complete, the logic matrices are tested from the train A and train B logic rack test. This step provides overlap between the analog and logic portions of the test program. During this test, all of the logic inputs are actuated automatically in all combinations of trip and non-trip logic. Trip logic is not maintained sufficiently long enough to

General Warning Alarm Reactor Trip

Each of the two trains of the solid state protection system is continuously monitored by the general warning alarm reactor trip subsystem. The warning circuits are actuated if undesireable train conditions are set up by improper alignment of testing systems, circuit malfunction or failure, etc. as listed below. A trouble condition in a logic train is indicated in the control room. However, if any one of the conditions exists in train A at the same time any one of the conditions exists in train B, the general warning alarm circuits will automatically trip the reactor.

ICSB Q6

a) Loss of either of two 48 volt dc or either of two 15 volt dc power supplies.

b) Printed circuit card improperly inserted.

c) Input Error Inhibit switch in the INHIBIT position.

d) Slave relay tester Mode Selector in TEST position.

e) Multiplexing selector switch in INHIBIT position.

f) Opposite train bypass breaker racked in and closed.

g) Permissive or Memory test switch not in OFF position.

h) Logic Function test switch not in OFF position.

TABLE 7.2.2-1

REACTOR TRIP CORRELATION

ICSB Q98

TRIP[a]	ACCIDENT[b]	TECH SPEC.[c] (Chapter 16)
1. Power Range High Neutron Flux Trip (Low Setpoint)	1. Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Condition (15.4.1)	2.2.1 Table 2.2-1 #2
	2. Excessive Heat Removal Due to Feedwater System Malfunctions (15.1.1)	
	3. Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection) (15.4.8)	
2. Power Range High Neutron Flux Trip (High Setpoint)	1. Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Condition (15.4.1)	2.2.1 Table 2.2-1 #2
	2. Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (15.4.2)	
	3. Startup of an Inactive Reactor Coolant Loop (15.4.4)	
	4. Excessive Heat Removal Due to Feedwater System Malfunctions (15.1.1)	
	5. Excessive Load Increase Incident (15.1.3)	
	6. Accidental Depressurization of the Main Steam System (15.1.4)	
	7. Major Secondary System Pipe Ruptures (15.1.5)	
	8. Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection) (15.4.8)	

TABLE 7.2.2-1 (Continued)

REACTOR TRIP CORRELATION

ICSB Q98

TRIP[a]	ACCIDENT[b]	TECH SPEC.[c] (Chapter 16)
3. Intermediate Range High Neutron Flux Trip	1. Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Condition (15.4.1)	See Note d 2.2.1 Table 2.2-1 #5
4. Source Range High Neutron Flux Trip	1. Uncontrolled Rod Cluster Control Assembly Bank Withdrawal From a Subcritical Condition (15.4.1)	See Note d 2.2.1 Table 2.2-1 #6
5. Power Range High Positive Neutron Flux Rate Trip	1. Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection) (15.4.8)	2.2.1 Table 2.2-1 #3
6. Power Range High Negative Flux Rate Trip	1. Rod Cluster Control Assembly Misaligment (15.4.3)	2.2.1 Table 2.2-1 #4
7. Overtemperature ΔT Trip	1. Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (15.4.2)	2.2.1 Table 2.2-1
	2. Uncontrolled Boron Dilution (15.4.6)	
	3. Loss of External Electrical Load and/or Turbine Trip (15.2.2/15.2.3)	
	4. Excessive Heat Removal Due to Feedwater System Malfunctions (15.1.1)	
	5. Excessive Load Increase Inciɑent (15.1.3)	
	6. Accidental Depressurization of the Reactor Coolant System (15.6.1)	
	7. Accidental Depressurization of the Main Steam System (15.1.4)	

REACTOR TRIP CORRELATION

	TRIP[a]	ACCIDENT[b]	TECH SPEC.[c] (Chapter 16)
		8. Loss of Reactor Coolant From Small Ruptured Pipes or From Cracks in Large Pipes Which Actuates ECCS (15.6.5)	
ICSB Q98	8. Overpower ΔT Trip	1. Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (15.4.2)	2.2.1 Table 2.2-1
		2. Excessive Heat Removal Due to Feedwater System Malfunctions (15.1.1)	
		3. Excessive Load Increase Incident (15.1.3)	
		4. Accidental Depressurization of the Main Steam System (15.1.4)	
		5. Major Secondary System Pipe Ruptures (15.1.5)	
	9. Pressurizer Low Pressure Trip	1. Accidental Depressurization of the Reactor Coolant System (15.6.1)	2.2.1 Table 2.2-1 #9
		2. Loss of Reactor Coolant From Small Ruptured Pipes or From Cracks in Large Pipes Which Actuates ECCS (15.6.5)	
		3. Major Reactor Coolant System Pipe Ruptures (LOCA) (15.6.5)	
		4. Steam Generator Tube Rupture (15.6.3)	
	10. Pressurizer High Pressure Trip	1. Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (15.4.2)	2.2.1 Table 2.2-1 #10
		2. Loss of External Electrical Load and/or Turbine Trip (15.2.2/15.2.3)	
	11. Pressurizer High Water Level Trip	1. Uncontrolled Rod Cluster Control Assembly Bank at Power (15.4.2)	2.2.1 Table 2.2-1 #11

REACTOR TRIP CORRELATION

ICSB Q98

TRIP[a]		ACCIDENT[b]		TECH SPEC.[c] (Chapter 16)
		2.	Loss of External Electrical Load and/or Turbine Trip (15.2.2/15.2.3)	
12.	Low Reactor Coolant Flow	1.	Partial Loss of Forced Reactor Coolant Flow (15.3.1)	2.2.1 Table 2.2-1 #12
		2.	Loss of Offsite Power to the Station Auxiliaries (Station Blackout) (15.2.6)	
		3.	Complete Loss of Forced Reactor Coolant Flow (15.3.1)	
13.	Reactor Coolant Pump Under-voltage Trip	1.	Complete Loss of Forced Reactor Coolant Flow (15.3.1)	2.2.1 Table 2.2-1 #15
14.	Reactor Coolant Pump Under-frequency Trip	1.	Complete Loss of Forced Reactor Coolant Flow (15.3.1)	2.2.1 Table 2.2-1 #16
15.	Low-low Steam Generator Water Level Trip	1.	Loss of Normal Feedwater (15.2.7)	2.2.1 Table 2.2-1 #13
16.	Safety Injection Signal Actuation Trip	1.	Accidental Depressurization of the Main Steam System (15.1.4)	See Note e
17.	Reactor Trip on Turbine Trip	1.	Loss of External Electrical Load and/or Turbine Trip (15.2.2/15.2.3)	See Note d 2.2.1 Table 2.2-1
		2.	Loss of Offsite Power to Station Auxiliaries (Station Blackout) (15.2.6)	See Note d 2.2.1 Table 2.2-1
18.	Manual Trip		Available for all Accidents (Chapter 15)	See Note d

7.3.1.2.4 Limits, Margins and Levels

Prudent operational limits, available margins, and setpoints before onset of unsafe conditions requiring protective action are discussed in Chapters 15 and 16.

7.3.1.2.5 Abnormal Events

The malfunctions, accidents, or other unusual events which could physically damage protection system components or could cause environmental changes are as follows:

1. Loss of coolant accident (See Chapter 15)
2. Steam breaks (See Chapter 15)
3. Earthquakes (See Chapters 2 and 3)
4. Fire (Section 9.5.1)
5. Explosion (Hydrogen buildup inside containment) (See Section 15.4)
6. Missiles (See Section 3.5)
7. Flood (See Chapters 2 and 3)

7.3.1.2.6 Minimum Performance Requirements

Minimum performance requirements are as follows:

1. System Response Times

ICSB Q29

The ESFAS response time is defined as the interval required for the ESF sequence to be initiated subsequent to the time that the appropriate variable(s) exceed this setpoint(s). The ESF sequence is initiated by the output of the ESFAS which is by the operation of the dry contacts of the slave relays (600 series relays) in the output cabinets of the solid state protection system. The response times listed below include the interval of time which will elapse between the time the parameter as sensed by the sensor exceeds the safety setpoint and the time the solid state protection system slave relay dry contacts are operated. These valves (as listed below) are maximum allowable values consistent with the safety analyses and the Technical Specifications and are systematically verified during plant preoperational startup tests. For the overall ESF response time; refer to Table 3.3-5 of the Technical Specifications. In a similar manner for the overall reactor trip system instrumentation response time; refer to Table 3.3-2 of the Technical Specifications.

The Engineered Safety Features Actuation System is always capable of having response time tests performed using the same methods as those tests performed during the preoperational test program or following significant component changes.

Much longer times are associated with the actuation of the mechanical and fluid system equipment associated with Engineered Safety Features. This includes the time required for switching, bringing pumps and other equipment to speed, and the time required for them to take load.

Operating procedures require that the complete Engineered Safety Features Actuation System normally be operable. However, redundancy of system components is such that the system operability assumed for the safety analyses can still be met with certain instrumentation channels out of service. Channels that are out of service are to be placed in the tripped mode or bypass mode in the case of containment spray.

7.3.2.4.1 Loss of Coolant Protection

By analysis of loss of coolant accident and in system tests it has been verified that except for very small coolant system breaks which can be protected against by the charging pumps followed by an orderly shutdown, the effects of various loss of coolant accidents are reliably detected by the low pressurizer pressure
ICSB Q70 signal; the Emergency Core Cooling System is actuated in time to prevent or limit core damage.

For large coolant system breaks the passive accumulators, including the UHI accumulators inject first, because of the rapid pressure drop. This protects the reactor during the unavoidable delay associated with actuating the active Emergency Core Cooling System phase.

High containment pressure also actuates the Emergency Core Cooling System. Therefore, emergency core cooling actuation can be brought about by sensing this other parameter which is a direct consequence of a primary system break. Thus, the Engineered Safety Features Actuation System detects the leakage of the coolant into the Containment. The generation time of the actuation signal of approximately 1.5 second, after detection of the consequences of the accident, is adequate. Containment spray will provide additional emergency cooling of containment and also limit fission product release upon sensing elevated containment pressure (high-high) to mitigate the effects of a loss-of-coolant accident.

The delay time between detection of the accident condition and the generation of the actuation signal for these systems is assumed to be 1.0 second; well within the capability of the protection system equipment. However, this time is short compared to that required for startup of the fluid systems.

The analyses in Chapter 15 show that the diverse methods of detecting the accident condition and the time for generation of the signals by the protection systems are adequate to provide reliable and timely protection against the effects of loss-of-coolant.

7.3.2.4.2 Steam Line Break Protection

The Emergency Core Cooling System is also actuated in order to protect against a steam line break. Approximately 2.0 seconds elapse between the sensing of

ANSI N210-1976, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations" Section 5.1.12.1 and 5.1.12.2 when a full loading of the assemblies described in Chapter 4 is considered. The computer codes and techniques utilized in the analysis have been validated against experimental data for water moderated UO_2 lattices with characteristics similar to the fuel analyzed.

In the analysis, the new fuel assemblies are assumed both to be in an infinite array and in their most reactive condition, namely unirradiated with 3.5 wt. percent enrichment U-235 and no control rods or supplemental neutron poisons present. All parameters are chosen to maximize K_{eff}, and the effects of reflectors other than water are included if their neglect would have been non-conservative.

Q410.10 The design of normally dry new fuel storage racks is such that the effective multiplication factor does not exceed 0.98 with fuel of the highest anticipated enrichment in place, assuming optimum moderation (under dry, fogged, or flooded conditions). For the fully flooded condition assuming cold, clean, unborated water, the value of K_{eff} is less than or equal to 0.95. Credit is taken for the inherent neutron-absorbing effect of the materials of construction.

The analysis concludes that a criticality accident during refueling and storage is not considered credible.

Since each unit has its own independent New Fuel Storage Facility and related racks, there are no safety considerations related to sharing of components.

Analysis and design of the New Fuel Storage Buildings and new fuel storage racks are performed as stated in Section 3.8.4. Details of the seismic analysis and design are provided in Section 3.7. Governing codes for design are as stated in Table 3.8.1-1.

9.1.2 SPENT FUEL STORAGE

9.1.2.1 Design Bases

Conformance with Regulatory Guide 1.13, "Fuel Storage Facility Design Basis" is as follows:

(1) Regulatory Position 1

Q220.45 The spent fuel storage facility including the spent fuel storage racks and the spent fuel pool liner plate, as part of the Auxiliary Building, is analyzed and designed as a Category I structure (see Table 3.2.1-1). For details of the loading conditions and loading combinations of the spent fuel pool, refer to Table 3.8.1-2.

(2) Regulatory Position 2

(a) Tornado winds are discussed in Section 3.3.2 and tornado missiles in Section 3.5.1.4.

(b) The spent fuel pool superstructure and the New Fuel Building together provide protection from normal and tornado winds, and prevent tornado generated missiles from contacting fuel within the pool.

Two isolation valves are provided on each outside air intake to ensure closure of the intake on high radiation level high chlorine concentration or high smoke concentration.

During normal operation each of the two 100% capacity, redundant outside air intakes are used to handle approximately 2000 cfm each. Each intake is sized to handle the 4000 cfm required for pressurization of the control room and control room area.

Q410.23 Upon detection of high radiation, high chlorine or smoke concentration, the affected intake's isolation valves close automatically. The outside air intakes are physically separated by approximately 420 feet and are shielded from site dispersion patterns by the Reactor Building structures. In the unlikely event both intakes are isolated, pressurization air no longer available; therefore, long-term pressurization following complete isolation is not possible. In order to restore pressurization, station procedures instruct the operator to select the desired intake based on inspection of control room indicators. During control room isolation, additional recirculation flow would be forced through the pressurizing filter train.

During a postulated period when both intakes are closed, assuming a 1/32 in. crack around all duct openings, cable openings, and door openings due to aging of the sealant, a 1/8 in. water gauge pressure differential would cause a 1125 cfm leak rate. A pressure differential of 0.05 in. water gauge, which would be a more realistic figure, would cause a 710 cfm leak rate. This is based on the fact that architectural design eliminates outside effects such as wind, etc.

9.4.1.4 Inspection and Testing Requirements

The Control Room Area Ventilation System is in continuous operation and is accessible for periodic inspection. Essential electrical components, switchovers, and starting controls are tested during preoperational tests and periodically thereafter to demonstrate system readiness and operability and as required by the Technical Specifications.

9.4.2 FUEL HANDLING AREA VENTILATION SYSTEM

9.4.2.1 Design Bases

The Fuel Handling Area Ventilation System is designed to: (1) maintain a suitable environment for the operation, maintenance, and testing of equipment; (2) maintain a suitable environment for personnel access; and (3) maintain the fuel handling and storage building at a negative pressure relative to the atmosphere to minimize outleakage as indicated on Figures 9.4.2-1, -2, and -3.

The Fuel Handling Area Ventilation System is designed to maintain a maximum inside temperature to 110°F and a minimum inside temperature to 50°F. Outdoor design temperatures meet or exceed those given in Table No. 1, Chapter 23 of the ASHRAE 1977 Fundamentals Handbook.

12.3.3.4 Typical System

The Auxiliary Building Ventilation System is used as an illustrative example of a typical air clean-up system design.

The Auxiliary Building general ventilation supply subsystem has a major impact on the personnel protection features incorporated into the design of the ventilation system. To control airborne activity, the Auxiliary Building ventilation supply air is delivered to the "clean" areas and areas of general personnel occupancy. This air is then routed to areas of greater contamination potential by pressure gradients induced by the exhaust system. Air is supplied and exhausted from the various areas of the Auxiliary Building as shown in Figures 9.4.3-1, -2, -3 and -4. As shown in Figures 9.4.3-1, -2, -3 and -4, some of the potentially contaminated areas of the Auxiliary Building require a direct supply of clean air in order to maintain the desired environment. However, in all such cases the quantity of air exhausted exceeds the amount supplied directly, thus confining any airborne contaminants to the subject space(s).

Clean outside air is supplied to areas of the Auxiliary Building through a bank of 2" deep prefilters and a bank of bag filters with an average efficiency of not less than 45% by the NBS test method for atmospheric dust. Each filter bank is provided with static pressure taps and an indicating gauge to measure the pressure drop across the filters. Filter change out is conducted in accordance with the manufacturer's recommendations. All exhaust from contaminated areas is exhausted through filter trains, all of which contain, as a minimum, prefilters, HEPA filters, and a charcoal adsorber in that order. In addition, all filter trains which are classified as Nuclear Safety-Related have a moisture eliminator and preheater upstream of the prefilters and another bank of HEPA filters downstream of the charcoal absorber. A layout of a typical safety-related filter train is illustrated in Figure 12.3.3-1. Prefilters, moisture eliminators, and HEPA filters are provided with static pressure taps and indicating gauges to measure the pressure drop across the filters. Filter change-out is conducted in accordance with the filter train manufacturer's recommendations. Testing and changing of charcoal is conducted in accordance with ANSI N510-1975.

The design of the Nuclear Safety-Related filter systems has been compared with the 1978 edition of Regulatory Guide 1.52. This comparison can be found in Tables 12.3.3-1, -2, -3 and -4.

A description of the ESF Filter System design parameters is given in Table 12.3.3-5.

12.3.4 AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING INSTRUMENTATION

12.3.4.1 Area Radiation Monitoring System

The Area Radiation Monitoring System is provided to monitor radiation levels in various plant locations that are potential radiation exposure areas. Indications and alarms from this system are used as an aid in conjunction with information from process radiation monitors (Section 11.5), plant operating procedures, and administrative controls to assure that station personnel exposure remains as low as reasonably achievable within 10CFR20 limits. Additionally, the Area Radiation Monitoring System assists in compliance with General

TABLE 12.3.3-5

ESF FILTER SYSTEM DESIGN PARAMETERS

SYSTEM	CARBON BED (MINIMUM DEPTH, IN.)	HEATER SIZE (KW)
Annulus Ventilation	2	45
Control Room Area Pressurizing	4	25
Fuel Handling Area Exhaust	2	80
Auxiliary Building Filtered Exhaust	2	30

123.0 MATERIALS ENGINEERING BRANCH--COMPONENT INTEGRITY SECTION

123.1 Indicate whether the individuals performing the fracture toughness tests are qualified by training and experience and whether their competency was demonstrated in accordance with a written procedure. If the above information cannot be provided, state why the information cannot be provided and identify why the method used for qualifying individuals is equivalent to those of Paragraph III.B.4 Appendix G, 10 CFR Part 50.

Response:

Although formal training programs were not in effect at all of the primary pressure boundary component vendors at the time of the manufacture of the Catawba components, the training of personnel engaged in testing was emphasized by all vendors. The training, plus the on-the-job experience in performing these routine tests, provides assurance that these tests were correctly performed.

123.2 To demonstrate compliance with the fracture toughness requirements of Paragraph IV.A.1 of Appendix G, 10 CFR Part 50:

a. Provide the RT_{NDT} for all RCPB welds which may be limiting for operation of the reactor vessels.

b. Indicate whether there are any RCPB heat-affected zones which require CVN impact testing per paragraph NB-4335.2 of the 1977 ASME Code. Provide CVN impact test data for these heat-affected zones which may be limiting for operation of the reactor vessels.

c. Indicate that there are no ferritic RCPB base metals other than in vessels which require fracture toughness testing to NB-2300 of the ASME Code. If there are ferritic RCPB base metals other than in vessels which require fracture toughness testing to NB-2300 of the ASME Code, provide CVN impact and drop weight data for all materials which will be limiting for operation of the reactor vessels.

Response:

a. The limiting RCPB material is that material which has the highest RT_{NDT} at end of life. The limiting material is not weld material for the Catawba units. The limiting material for Catawba Unit 1 is the RV intermediate shell forging, as identified in Table 5.3.1-4. The limiting material for Unit 2 is intermediate shell plate B8605-1, as identified in Table 5.3.1-5.

b. Charpy V-notch test data for the heat affected zone of the limiting beltline region plate or forging is presented in the surveillance program report for each unit. (Unit 1 report: WCAP-9734; Unit 2 report: available May 1984.) No other heat affected zone material requires impact testing per NB-4335.2 of the 1977 ASME Code.

c. There are no ferrite base metals other than in vessels in the reactor coolant pressure boundary.

123.3 Provide material test data, analysis or data from the literature to demonstrate that bolts from heat number 35674 which had CVN values of 29 to 38 ft-lbs and 8 to 12 mils LE at 10°F, would meet the requirements of NB-2333 of the ASME Code (45 ft-lbs and 25 mils L.E. at the preload or lowest service metal temperature which ever is less). The sample material which demonstrates that the heat no. 35674 bolts will comply with the CVN requirements of NB-2333 of the ASME Code, must have been fabricated to an equivalent material specification and heat treated to an equivalent metallurgical condition as the material from heat no. 35674 bolts.

Response:

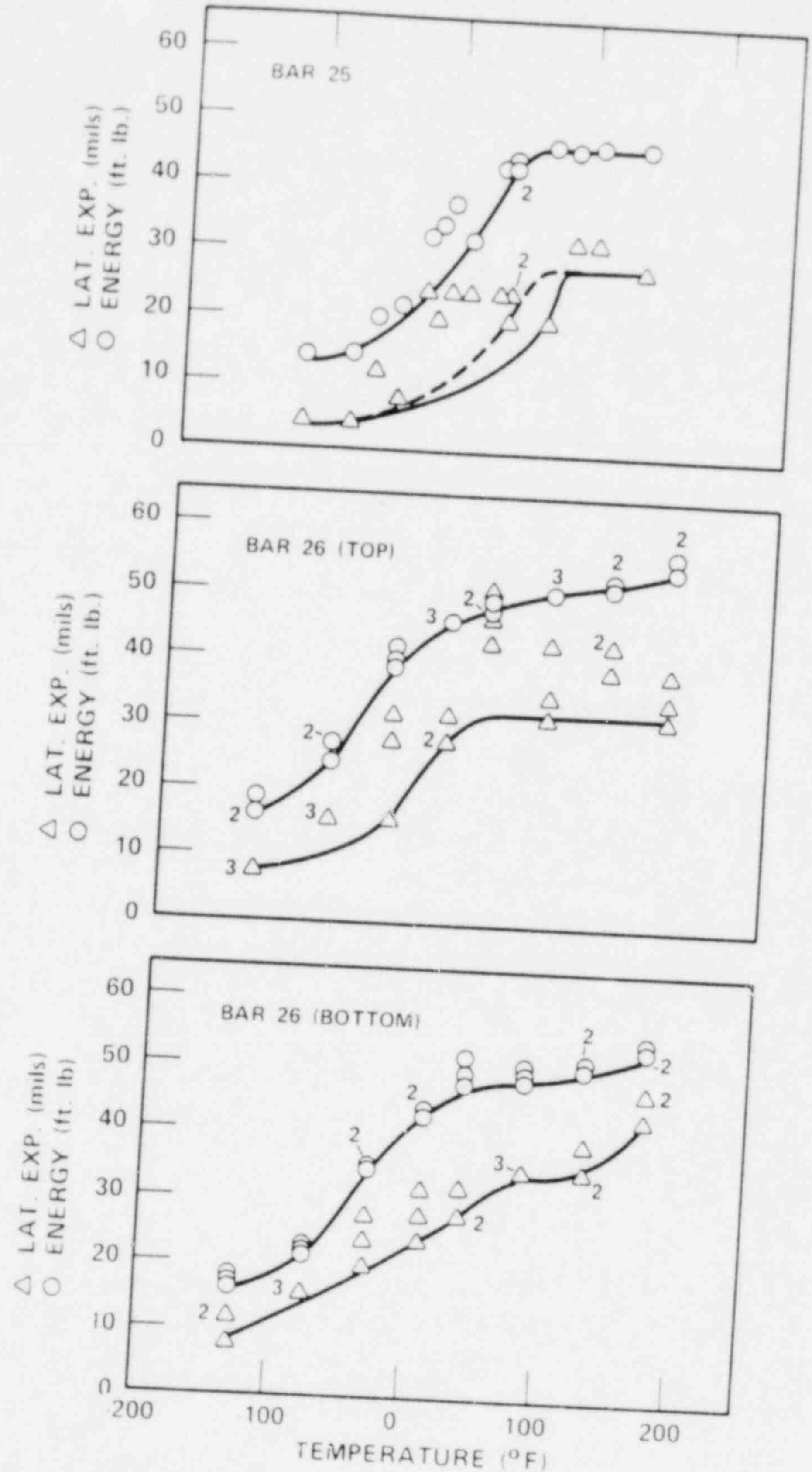
The Catawba Unit 1 reactor vessel was designed in accordance with the 1971 Edition, Winter 1971 Addendum of the ASME Code. The fracture toughness requirement for the vessel closure studs material is 35 foot pounds Charpy energy at a temperature 60°F below the lowest service temperature.

The Charpy data shown for Bar 25T in Table 5.3.1-6 has been changed to correct a typographical error. The 29 foot pounds shown originally should have been 39 foot pounds. Additional Charpy data from heat number 35674 is shown in Table Q123.3-1. This same data is shown graphically in Figure Q123.3-1. One of these 128 data points (MLE for Bar 25 at 68°F) appears uncharacteristically low. Ignoring this one point, the curves of Figure Q123.3-1 demonstrate that this material would satisfy the requirement of later editions of the ASME Code for 45 foot pounds and 25 MLE at temperatures as low as 50°F.

TABLE Q123.3-1

CHARPY V-NOTCH IMPACT DATA FOR CLOSURE STUD MATERIAL FROM HEAT NO. 35674

BAR 25				BAR 26 (TOP)				BAR 26 (BOTTOM)			
TEMP (°F)	ENERGY (ft/lb)	LAT. EXP. (MILS)	SHEAR (%)	TEMP (°F)	ENERGY (FT/LB)	LAT. EXP. (MILS)	SHEAR (%)	TEMP (°F)	ENERGY (FT/LB)	LAT. EXP. (MILS)	SHEAR (%)
-112	14.0	4	26	-130	16.5	8	20	-130	17.5	12	21
-76	14.5	4	25	-130	16.5	8	27	-130	18.0	8	16
-58	20.0	12	39	-130	19.0	8	21	-130	16.5	12	16
-40	22.0	8	43	-76	24.5	16	48	-76	23.0	16	30
-22	32.5	24	47	-76	27.5	16	30	-76	22.5	16	20
-13	34.0	20	51	-76	27.5	16	34	-76	22.0	16	40
-4	37.5	24	47	-31	39.5	32	39	-31	35.0	24	34
10	32.0	24	55	-31	40.0	16	59	-31	35.5	20	40
32	43.0	24	55	-31	42.0	28	47	-31	35.5	28	40
40	44.0	24	55	10	46.0	32	43	10	44.0	32	55
40	43.0	24	55	10	46.0	28	39	10	44.0	24	44
40	43.0	20	51	10	46.0	28	43	10	43.0	28	43
68	46.5	20	55	40	49.5	43	44	40	52.0	28	43
86	46.0	32	55	40	48.0	47	47	40	49.5	28	43
104	46.5	32	55	40	48.0	51	51	40	48.0	32	34
140	46.5	28	55	86	50.5	32	39	86	49.0	35	51
				86	50.5	35	43	86	49.5	35	51
				86	50.5	39	43	86	50.5	35	47
				131	52.5	39	39	131	50.5	35	47
				131	52.5	43	43	131	51.5	35	55
				131	52.0	43	43	131	51.5	39	47
				176	56.5	39	51	176	54.5	43	43
				176	56.5	32	55	176	54.0	47	35
				176	54.5	35	55	176	54.0	47	39



CHARPY V-NOTCH IMPACT DATA
FOR CLOSURE STUD MATERIAL
FROM HEAT NO. 35674

CATAWBA NUCLEAR STATION
Figure 123.3-1
Rev. 6

123.4 Provide material test data, analysis or data from the literature which demonstrates that the intermediate to lower shell weld (P710) has an RT_{NDT} of 0°F and an upper shelf greater than 75 ft-lbs.

The additional data should be from similar welds, i.e., those having the same type of weld wire and flux and thermal treatment as weld (P710). The information should include a comparision of the significant weld parameters (e.g., weld wire, flux and thermal treatment) and mechanical properties from the sample and (P710) beltline weld.

Response:

As shown in Table 5.3.1-2, weld P710 uses the same wire and flux type as weld R747. These welds were fabricated by the same vendor using the same processes and heat treatments. The fracture toughness data verses temperature for weld R747 are shown in Table 5.3.3-4. Also shown in that Table are the data for weld P710 at 10°F. This information supports the estimates of an RT_{NDT} of 0°F and an upper shelf of greater than 75 foot pounds for weld P710.

123.5 Provide actual pressure-temperature limits for CNS-1 and 2 based upon the limiting fracture toughness of the reactor vessel material and the predicted shift in the adjusted reference temperature, RT_{NDT} resulting from radiation damage. The pressure-temperature limits for the following conditions must be included in the technical specifications when they are submitted:

1. Preservice hydrostatic tests,

2. Inservice leak and hydrostatic tests,

3. Heatup and cooldown operations, and

4. Core operation.

Response:

The minimum preservice hydrostatic test temperature for both Units 1 and 2 is 150°F. The pressure-temperature limits for other modes of operation will be included in the Technical Specifications.

123.6 Provide full CVN impact curves for each weld and plate in the beltline region. Provide the data in tabulated and graphical form.

Response:

The fracture toughness test details for the reactor vessel beltline materials are shown in table 5.3.3-4 for Unit 1 and Table 5.3.3-5 for Unit 2.

123.7 To demonstrate the surveillance capsule program complies with Paragraphs II.B and II.C.3 of Appendix H.

a. Provide the withdrawal schedule for each capsule.

b. Provide the lead factors for each capsule.

c. Provide a sketch which indicates the azimuthal location for each capsule relative to the reactor core.

Response:

Details regarding the surveillance capsule program for Unit 1 are provided in WCAP-9734. A similar report will be available for Unit 2 in approximately May 1984.

safety-related systems functions. Relate your description to s ific drawings you have made or will make available.

Also indicate the location of air intakes and exhausts on drawi s and describe missile and tornado protection for safety related equipment near these openings. Provide drawings of the auxiliary shutdown panel room ventilation system.

Response:

Discussions of the Auxiliary Building Ventilation systems are provided in Section 9.4.3.2, and a discussion of the Fuel Building Ventilation system is provided in Section 9.4.2.3.

Reference Section 9.2.1 description of continuous cooling provided for safety-related pumps. The following pumps are provided with water-cooled motors and where necessary, bearing coolers:

1. Safety Injection Pumps (NI System)
2. Residual Heat Removal Pumps (ND System)
3. Centrifugal Charging Pumps (NV System)
4. Containment Spray Pumps (NS System)
5. Auxiliary Feedwater Pumps (CA System)

Loss of offsite power and the failure of a single active component constitute a plant mild environment condition. The issue of mild environments is currently being pursued by the NRC and by industry. Duke Power Company is participating in the industry effort via EPRI and AIF.

Protection of air intakes and exhausts is discussed in Section 3.5.2.

The Auxiliary Shutdown Panel Rooms Air Conditioning System is shown on Figures 9.4.3-2, 9.4.3-6, and 9.4.3-7.

410.26 (9.4.5) Describe the ventilation system for the nuclear service water pump house and verify that a proper environment is maintained for safety related components within the pump house assuming a loss of offsite power and any concurrent single active failure.

Response:

See revised Section 9.4.8.3.

A-17 - Systems Interaction

The design of Catawba Nuclear Station is based on the principle of defense-in-depth. Implementation of this design principle results in physically separated redundant safety systems which are protected against events such as high energy line ruptures, missiles, high winds, flooding, seismic events, fires and sabotage.

Essential equipment is located inside Category I structures which are designed to protect the equipment against external environmental effects and missiles. The redundant safety systems are designed to be both physically separate and independent. Non-safety equipment is either seismically mounted/restrained or located so as to prevent adverse interactions with safety equipment during seismic events.

The Catawba Nuclear Station is reviewed for pipe rupture events, non-seismic piping interactions during a seismic event, and internal flooding as a result of fluid released from either of the two aforementioned postulated occurrences. If any unacceptable interaction is encountered, protection is provided in the form of equipment/pipe/cable relocation or the installation of intervening or restraining devices.

In response to NRC Branch Technical Position APCSB 9.5-1, Appendix A (Fire Hazard Analysis), Duke has addressed the adequacy of the fire protection program for meeting the single failure criterion and physical separation requirements (such as block walls) as well as the interaction of fire supression equipment on other systems. It is our conclusion that system interactions are adequately addressed and that protection against system interactions either has been or will be provided through the commitments in the Fire Hazards Analysis.

In reference to sabotage, Duke has conducted a preliminary vital area analysis to assure the protection of required Engineered Safety Features (ESF) for McGuire and Catawba Nuclear Stations and from this analysis developed the plant security systems. Since the ESF and security system design at Catawba and McGuire are essentially identical and the sabotage protection is in place and approved for McGuire, it is Duke's position that the level of sabotage protection provided at Catawba is acceptable.

A response was developed by Duke for control system interaction and is documented in A-47 (Safety Implications of Control Systems).

The aforementioned analyses and design features adequately address generic issue A-17, System Interaction. Therefore, we conclude that the Catawba Nuclear Station can be operated prior to resolution of this generic issue without undue risk to the health and safety of the public.

810.0 EMERGENCY PLANNING

810.1 (13.3) (AR) As per Regulatory Guide 1.70, provide plots showing ground-level doses for stationary individuals. For both whole body and thyroid resulting from the most serious DBA.

Response:

The Emergency Plan for Catawba is being revised to reflect the results of recent NRC reviews of the Oconee and McGuire Emergency Plans.

810.2 (13.3) (AR) Discuss the extent of compliance of your emergency planning with NUREG-0696.

Response:

Duke Power Company's Crisis Management Plan and Catawba Nuclear Station's Emergency Plan, Section H - "Emergency Facilities and Equipment," describe the Crisis Management Center - CMC (i.e., EOF), Technical Support Center (TSC), and the Operational Support Center (OSC) for Catawba. These facilities, when completed, will meet the intent of review criteria established in NUREG-0654, FEMA-REP-1, Rev. 1 and NUREG-0696.

Emergency Response Facility (ERF) conceptual design information is provided in the following areas which allow a complete discussion of the extent of compliance with NUREG-0696:

Facility description including communications, monitoring capability, data acquisition, information availability, habitability, size, location, SPDS, Nuclear Data Link, function, location, staffing and training, size, and structure.

Catawba Nuclear Station - Emergency Response Facilities

1. Approach

Duke Power Company's approach to establishment of ERFs at Catawba was the same as that taken at its Oconee and McGuire Nuclear Stations. The company's plans center around maximum use of existing facilities, with modifications and expansions where necessary.

This approach was chosen since it would lead to early operability of our emergency facilities and would take advantage of the large instrumentation data base available through existing computer systems. These computer systems are part of a family of similar systems which have been successfully applied in fossil, hydro

and nuclear plant installations since 1963. Total availability of these systems, even those installed in the 1960's, averages around 99%, even considering computer outages during periods of no need (unit outages, etc.). Furthermore, no additional training will be required of plant operators since they are already familiar with existing computer systems (no new sources of confusion will be introduced into the control room).

This approach is a part of the overall corporate Crisis Management Plan, which involves all appropriate levels of corporate management.

2. Technical Support Center (TSC)

Duke Power Company's plans for a TSC at Catawba are described in Section H of the Crisis Management and Station Emergency Plans. The TSC is located in a four-bay area between column lines 25-27, S-V on elevation 594 in the Service Building. The TSC is within two minutes walking distance from the Control Room.

The TSC is utilized for evaluation of plant status by knowledgeable personnel in support of operations during an emergency situation. This center will be used to direct the onsite and initial offsite aspects of an emergency. Anticipated occupants are the Station Manager, Station Superintendents, station staff (Health Physics, Chemistry, Performance, I&E, maintenance, operations engineering), the NSSS supplier, and the Nuclear Regulatory Commission. Space is provided for up to 25 persons.

The TSC is capable of activation and full operation within one hour of an alert condition. The ability to activate and bring the TSC to a fully operational state will be tested each year during the annual emergency preparedness exercise.

The TSC is capable of withstanding the most adverse conditions reasonably expected within the design life of the plant including earthquakes, high winds (other than tornadoes), and floods.

The TSC has the same radiological habitability as the control room under accident conditions. Installed and portable radiation monitoring equipment will be in use in the TSC to indicate dose rates and airborne radioactivity concentrations in an emergency. Protective equipment including anticontamination clothing, respiratory protective gear and potassium iodide will be available. If the TSC becomes uninhabitable, the TSC management function will move to the control room area.

The TSC will have redundant two-way communication with the Control Room, the CMC, and the NRC Operations Center. Section F of the Station Emergency Plan describes the communications system for the TSC. There will be phone lines in the TSC for use day to

day during normal plant operations. These will be supplemented by additional plant extensions available in the TSC as "dedicated" telephones. These "dedicated" phones with a backup emergency radio system "provide a means for TSC access to commerical communications services that bypass any local offsite or onsite telephone switching facilities that may be susceptible to loss of power during emergencies.

Two "dedicated" phones of those described above will be provided for the NRC when the TSC is activated.

The TSC will be able to display, printout or trend comprehensive data necessary to monitor reactor systems status and to evaluate plant system abnormalities; in-plant and off-site radiological parameters and meteorological parameters are also available. This capability is provided via each unit's Operator Aid Computer and is independent of control room actions. Capability to access and display thousands of parameters, individually or in groups, is provided. Duke's experience with similar process computer installations would lead to expectations of data availability in the TSC of over 99%.

Data vital to TSC functions will be stored on data sheets accumulated throughout the incident. Loss of power to the DAS will not affect vital TSC functions, as past and present data transmission, storage, display, and calculational means will still be available via these data sheets and within the plant computer system.

Any parameters which are input to the OAC (Operator Aid Computer) are available for callup.

A combination of strip charts, event recorders, and OAC (Operator Aid Computer) printouts provides sufficient data to analyze the incident form a pre- and post-event data aspect. Furthermore, many of the parameters in Regulatory Guide 1.97 are available in the OAC. However, all Regulatory Guide 1.97 parameters are not appropriate for display in the TSC.

Further information on instrumentation, data systems, and meteorological capabilities are described in part H of the Station Emergency Plan.

The Data System employed in the TSC for transmission of plant status data to users outside the TSC is that in use at Oconee and McGuire. This system involves acquisition of data from the Operator Aid Computer, loading that information periodically onto a floppy disc (~ every 15 minutes) and reading the information on the disc onto the Distributed Data Processing System in place at each station and the General Office. When the data

is verified by TSC personnal, it is released for use by those outside the TSC who have been trained in operation of the sytem. NRC Region II Emergency Response Team members and personnel from the Operations Center in Bethesda were trained on access to this data March 1, 1982. Members of the Crisis Management Team Technical Support Group coordinate outside requests for information other than that provided on the primary status sheet.

TSC personnel would use a telecopier for transmission of hard-copy information offsite should the primary mechanism described above be unavailable.

TSC personnel have ready access to Technical Specifications, operating procedures, FSAR, operating records, and up to date drawings. This information, if not in the TSC, is readily available in the master file area or other areas of the plant.

3. Operational Support Center (OSC)

The OSC for Catawba is described in part H of the Crisis Management and Station Emergency Plans. Table H-5 of the CMP shows the location of the OSC relative to the control room.

The OSC is the place designated for essential operations support personnel to report in an emergency situation. This center may be used as a briefing area for such personnel in preparation for work assignments. The center will serve to reduce noise and confusion in the control room.

The OSC will have adequate "dedicated" and normal telephone communications capabilities.

4. Emergency Operations Facility - Crisis Management Center (CMC)

Portions of the Administration Building, Temporary Administration Building, and the Construction Training Building will comprise the CMC and Crisis News Center (CNC) for Catawba. Tables H-6 thru H-9 of the CMP describe the layout of these facilities.

The function of the CMC is to direct and control all emergency and recovery activities related to the utility, with emphasis on the coordination of offsite activities such as dispatching mobile radiological monitoring teams, communications with local, state, and Federal Agencies, and coordination of corporate and other outside support.

The CMC is large enough to provide working space and facilities for at least 50 persons, including ten NRC personnel. Anticipated occupants are the Recovery Manager and his advisors and staff,

clerical support, crisis news representatives, and the Nuclear Regulatory Commission.

The CNC is within a short walking distance of the CMC and can house in excess of 100 persons.

Space restrictions are not as stringent in the EOF as the TSC and thus, some separate area may be available on an "as-needed" basis. Space within the CMC and "designated" communications will be provided for NRC representatives.

The EOF will be activated annually for exercises and any drill deemed necessary by management. Activation specifically for training outside of these situations is not planned.*

The TSC will perform the functions of the EOF until full operation is established. EOF staffing for alert or higher level emergency conditions will include resources from the Recovery Manager and each of the functional groups. The staffing for each level will be essentially the same. The EOF activation drills described will be a portion of the emergency preparedness exercises held each year.

The CMC is a substantial structure, providing significant shielding (protection factor > 50) from direct outside radiation and is "well-engineered" as described in the Uniform Building Code.

As the CMC is well within 10 miles of the reactors (several hundred yards) use of a backup CMC is planned. Should the CMC be rendered inhabitable by a large radioactivity release, CMC personnel would evacuate the primary CMC and go to the backup facilities at the Charlotte General Office located some 45 minutes to 1 hour away. The capabilities and layout of the backup facility are described in part H of the CMP. The backup facility's capabilities are very similar to the primary CMC in the areas of communication, data acquisition, data display, records and drawings availability, and personnel availability.

The CMC will have redundant two-way communications with the TSC, Control Room, NRC, and appropriate offsite support agencies (including local government).

The EOF communications system includes "dedicated phones" as in the TSC, a ringdown line to the State FEOC, ringdown lines on the HP and Emergency Network of the NRC, and backup radio to the counties and state. This system will meet the intent of the guidance in Section 4.6 of NUREG-0696.

The data system described for the TSC to provide date to persons outside the TSC is available in the CMC. This is the primary

means of data acquisition in the CMC. Section I of the CMP provides further detail on information that is available. The backup means for data coming into the CMC or for leaving this facility is telecopier.

Section H of the CMP describes equipment and material stored for use by the organization in an emergency. Other records are available in the backup CMC as well as in the Administration Building at Catawba. These records include, Technical Specifications, operating procedures, drawings, FSAR, Emergency Plan, Population distribution data, environs monitoring records, and personnel exposure histories. These records are available to CMC personnel in an emergency.

V. Safety Parameter Display System (SPDS)

A March 25, 1981 letter from William O. Parker, Jr. to Harold R. Denton describes the Company's position and plans for SPDS.

VI. Nuclear Data Link (NDL)

The NRC's stated basis of need for the NDL/offsite data transmission (to allow accident assessment and thus protect the health and safety of the public) was addressed in Duke Power Company's comments on NUREG-0696 transmitted to the Secretary of the Commission on September 30, 1980 in a letter from William O. Parker, Jr. (General Comment No. 3)

Duke Power Company's basic concern is with the transmission of real-time data to any offsite facility such as the EOF, NRC operations center, or the state EOC.

Duke's emergency planning is based upon carrying out the minute-to-minute operation of the plant from the control room, where those personnel most qualified for that duty are located. Where sufficient time is available, advice and support in continuing operations would come from plant personnel in the Technical Support Center. Control room and TSC personnel are the most familiar with the particular plant and are in the best position to effectively use real-time data.

Periodic snapshots of plant conditions for information or detailed analysis can be provided much more reliably, simply and economically by telephone, telecopier or other timely but non real-time means. Voice communications would certainly be used to obtain different or additional data or to confirm data. In any event, a system such as the NDL has a high potential to degrade existing plant instrumentation systems.

The transmission of unreviewed data, which could be in error, has the potential for causing confusion, incorrect recommendations for protective action, and in the end be inimical to the health and safety of the public.

There are several areas relative to the NDL where responsibilities are not yet clear. For example, NUREG-0654 states that the facility operator (licensee) has primary responsibility to make recommendations regarding public protective actions, whereas one of the stated purposes of the NDL is to facilitate NRC's assessment of effects of an accident upon the general public. (Note that real-time data is not required for such assessments). Another undefined responsibility is that of NDL site equipment maintenance. As a final question, does NRC assume full responsibility and liability when "in extreme cases, the NRC directs that certain operations be performed by the licensee at the nuclear plant?"

In summary, the NDL could adversly impact existing information available to the operator and could be a source of confusion leading to an erroneous decision in an emergency. In light of the above, the industry cost of up to $200 million for the NDL cannot be justified.

810.3 (13.3) (AR) As per Regulatory Guide 1.70, provide a map showing all roads available for vehicular evacuation of the exclusion area and environs extending at least 10 miles from the plant.

Response:

See Figure Q810.3-1.

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810.4 (13.3) (AR) As per Regulatory Guide 1.70, provide a map showing demographic data in 1-mile increments from the plant to the outer boundary of the proposed low population zone.

Response:

See response to Q810.1.

810.5 The following comments apply to the Catawba Nuclear Station Emergency Plan (Plan) dated August 1980, and identify, in parentheses, the applicable evaluation criteria of NUREG-0654, Rev. 1:

General

a. The South Carolina, North Carolina and applicable county emergency plans must be submitted for review.
b. The applicant's procedures which implement the Catawba Plan must be submitted for review.
c. The corporate Crises Management Plan must be submitted for review.

Response:

a. The responsible agencies are in the process of developing their respective emergency plans for the Catawba Nuclear Station and should have them completed by March 31, 1983 and ready for submittal by May 1983.
b. The Catawba Nuclear Station Emergency Plan Implementing Procedures are being written by the station staff and will be available for review later.
c. The Crisis Management Plan addresses Duke Power Company's plans for emergency response at Oconee, McGuire and Catawba. Copies of this plan were previously submitted in support of Oconee and McGuire Emergency Plans. The Crisis Management Plan was recently revised to include Catawba.

810.6 (A.3) Written agreements do not exist for all agencies/organizations identified in Section 5.3.2 of the Plan which are relied upon for support and services. Provide these agreements.

Response:

Written agreements are provided in Appendix 5 of Catawba Nuclear Station Emergency Plan Revision 1 as listed. Other agreements will be included as soon as they are received.

810.17 (E.1) The procedures for notification of response organizations should be provided for review.

Response:

Emergency Procedure EP/0/A/5000/06, Notification of Unusual Event, provides for notification of response organizations and will be submitted for review by August 1982.

810.18 (E.2) The procedures for alerting, notifying, and mobilizing personnel should be provided for review.

Response:

Emergency Procedures EP/0/A/5000/07 through EP/0/A/5000/09 provide for alerting, notifying and mobilizing personnel and will be submitted for review by August 1982.

810.19 (E.6) The Plan should provide for prompt alerting, notifying and instructing the public in a manner which meets the criteria of Appendix 3, NUREG-0654.

Response:

The alert and notification system for the Catawba Nuclear Station is being developed. Further information will be supplied on July 30, 1982.

810.20 (F.2) The Plan should provide for a communications link for fixed/mobile medical support facilities.

Response:

Revision 1 to Catawba Nuclear Station Emergency Plan, Section F.2, describes the methods of communication for medical support facilities.

810.21 (F.3) The Plan should address periodic testing of emergency communications systems.

Response:

Revision 1 to Catawba Nuclear Station Emergency Plan, Section F.e, describes the testing of emergency communication systems.

810.22 (G.1, G.2) The actual means utilized for dissemination of information to the public must be identified in the Plan. The brochure, etc., should meet the specified criteria and should be submitted for review.

Response:

The brochure for the Catawba Nuclear Station will be submitted for review by October 1, 1983.

810.23 (G.3, G.4, G.5) Neither the Plan nor the referenced Crises Management addresses interaction with the news media. Provide this information.

Response:

Revision 1 to Catawba Nuclear Station Emergency Plan Sections G.3 and G.4 describe the Crisis Management interaction with the news media.

810.24 (H.1) The TSC and OSC should conform to the criteria of NUREG-0696.

Response:

The Technical Support Center (TSC) will have a sufficient room for designated personnel plus redundant communication system to the Control Room, CMC, counties EOC's states and NRC in Washington. Habitability will be controlled to the same degree as the Control Room. The TSC is within two (2) minutes walking distance of the Control Room. Located in the TSC will be emergency supplies, communication, and Operator Aid Computer Aid (OAC) terminal for plant status reports. The Operations Support Center (OSC) is located close to the Control Room and has sufficient communications and supplies for emergency personnel. Further detail on the TSC is provided in the Catawba Nuclear Station Emergency Plan Revision 1, Section H.

810.25 (H.2) The EOF should conform to the criteria of NUREG-0696.

Response:

The Crisis Management Center (CMC or EOF) as described in Section H.2 of Catawba Nuclear Station Emergency Plan Revision 1 will have the required space and facilities for the direction and control of all emergency and recovery activities. Mobile emergency monitoring teams can be dispatched from the CMC while constant communications are maintained with local, state and federal agencies.

810.26 (H.8) Meteorological instrumentation and procedures should conform to the criteria of Appendix 2, NUREG-0654.

Response:

Meteorological instrumentation is being designed to conform to Appendix 2 of NUREG-0654, Revision 1 as described in CNS Emergency Plan Section H. Procedures applicable to this instrumentation will be developed and submitted for review prior to Unit 1 Startup.

810.27 (H.11) The emergency equipment/supplies identified in Appendix 10.4 should be expanded to include all of the cateogries of this criterion.

Response:

Catawba Nuclear Station Emergency Plan Revision 1 Sections H.0 and H.11 describe the location, contents, functional checks and calibration of emergency kits and the equipment and supplies they contain.

810.28 (I.1) The Plan should identify plant system and effluent parameter values which are characteristic of a spectrum of off-normal conditions and accidents.

Response:

Plant system and effluent parameter values that are characteristic of abnormal plant conditions and accidents are identified in Figures D-1, D-2, D-3, and D-4 of Catawba Nuclear Station Emergency Plan Revision 1. These values indicate which emergency classification is appropriate as described in Appendix 1 of NUREG 0654 Revision 1.

810.29 (I.2) Post-accident sampling capability and containment radiation monitoring should be addressed in the Plan.

Response:

Section I.2 of Catawba Nuclear Station Emergency Plan Revision 1 states that post-accident sampling equipment has been purchased and is being installed in compliance with NUREG-0578.

810.30 (I.3) The methods/techniques for determining the source term of a release and the magnitude of the release should be addressed in the Plan.

Response:

Station Procedures have been developed to determine the source term and magnitude of a release and are addressed in Section I.3.a/I.3.b of Catawba Nuclear Station Emergency Plan Revision 1.

810.31 (I.4) The relationship between effluent monitor readings and onsite and offsite exposures/contamination for meteorological conditions should be addressed.

Response:

Station procedures referenced in Section I.3.a/I.3.b of Catawba Nuclear Station Emergency Plan Revision 1 establish the relationship between effluent monitor readings and onsite/offsite exposures and contamination for various meterological conditions.

810.32 (I.5) The capability of acquiring and evaluating meteorological information which meets the criteria of Appendix 2, NUREG-0654 should be addressed.

Response:

Catawba Nuclear Station Emergency Plan Revision 1 Section 1.5 states that meterological information will be available at the Crisis Management Center, Technical Support Center and Control Room through the Operator Aid Computer and the VAX Computer and by direct telephone communication. Meteorological Information to be supplied to the NRC and state EOC will be by the ring-down telephone and by telephone during the initial and subsequent followup messages.

810.33 (I.6) The Plan should address a method for determining release rate/projected doses if instrumentation is offscale or inoperative.

Response:

Station Procedure HP/0/B/1009/06, Alternative Method for Determining Dose Rate Within the Reactor Building, is being developed to address this item. The procedure will be ready for review by July 1, 1982.

810.34 (I.10) The Plan should provide for relating measured parameters to dose rates for key isotopes and gross radioactivity measurements and for estimating integrated dose from actual and projected dose rates and comparing these estimates with PAGs.

Response:

Procedures HP/0/B/1009/12, 13, 14, & 15 relate measured parameter to dose rates for key isotopes and are used to project dose rates for comparison with PAG's. These procedures are being developed and will be ready for review by January 1, 1983.

810.35 (J.8) The time estimates provided in the Plan for evacuation within the plume exposure EPZ should be revised to satisfy the criteria of Appendix 4, NUREG-0654.

Response:

Time estimate studies for Catawba will be done in a manner similar to that used at Oconee and McGuire as soon as the states of N.C. and S.C. complete work on evacuation shelters and reception centers. This document will be submitted for review by October 1983.

810.36 (J.10) Maps have not been included in the Plan. Provide these maps to meet this criterion.

Response:

Maps will be included in the Catawba Nuclear Station Emergency Plan, upon completion of N.C. and S.C. evacuation sector assignments to a 10 mile EPZ.

810.37 (K.3) The Plan should address 24-hour/day capability to determine doses received by emergency workers and the reading of dosimeters and record keeping for emergency workers.

Response:

Section K.3 of Catawba Nuclear Station Emergency Plan Revision 1 addresses the 24 hour capability to determine doses received by emergency personnel and record keeping of doses received.

810.38 (K.5) Decontamination of supplies, instruments and equipment and waste disposal should be addressed in the Plan.

Response:

Section K.5.b of Catawba Nuclear Station Emergency Plan Revision 1 addressed decontamination of supplies, instruments and equipment and waste disposal.

810.39 (L.1) The Plan states that the Charlotte Memorial Hospital is the backup medical facility; however, no written agreement is included for this facility. Provide this agreement.

Response:

The agreement letter has been added to the CNS Emergency Plan and will be included in Revision 2 to the Plan.

810.40 (N.1) Periodic exercises should simulate an emergency which results in offsite radiological releases.

Response:

Section N.1.a of Catawba Nuclear Station Emergency Plan Revision 1 states that an annual exercise as described in 10CFR Part 50 Appendix E.

810.41 (N.2) Communication with Federal emergency response organizations and States within the ingestion pathway should be tested quarterly.

Response:

Revision 1 of Catawba Nuclear Station Emergency Plan Section N.2.a changed the communications drill with Federal and State organizations in the ingestion pathway to quarterly.

810.42 (N.4) The Plan should state that official observers from Federal, State and/or local governments will observe and evaluate the exercise.

Response:

Section N.4 of Catawba Nuclear Station Emergency Plan Revision 1 states that an exercise critique will be held as soon as possible after each exercise and the Crisis Management Plan Section N states the county, state and federal agencies will be involved on the "full scale" exercises, including the use of official observers to evaluate the exercise.

810.43 (O.3) The Plan should address that first aid training is equivalent to Red Cross Multi-Media.

Response:

Section O.3 of Catawba Nuclear Station Emergency Plan Revision 1 states that Red Cross Multi-Media training is given for first aid training.

Response:

Section P.5 of Catawba Nuclear Station Emergency Plan Revision 1 addressed the criteria of emergency plan distribution.

810.50 (P.6) The Plan should contain a list of supporting plans.

Response:

Figure P-1 of Catawba Nuclear Station Emergency Plan Revision 1 is a detailed listing of supporting plans.

810.51 (P.7) The Plan should identify the actual procedures required to implement the Plan.

Response:

Emergency Plan implementing procedures have been identified and will be provided in Revision 2 to the Plan.

810.52 (P.8) The cross-reference to NUREG-0654, which is included in the Plan should be accurate and specific.

Response:

Revision 1 of the Catawba Nuclear Station Emergency Plan is written in the format of NUREG 0654, Revision 1.

810.53 (P.9) An independent review of the emergency preparedness program is to be conducted every 12 months. Provide this commitment in the Plan.

Response:

The independent audit (review) of the Emergency Preparedness Program has been changed to every 12 months in Section P.9 of Catawba Nuclear Station Emergency Plan Revision 1.

810.54 (P.10) The Plan should address updating of telephone numbers in emergency procedures.

Response:

Section P.10 address updating of telephone numbers listed in Catawba Nuclear Station Emergency Plan Implementing Manual.