SNUPPS

Standardized Nuclear Unit
Power Plant System

5 Choke Cherry Road
Rockville, Maryland 20850
(301) 869-8010

March 20, 1984

SLNRC 84-0046 FILE: 0543
SUBJ: SNUPPS Technical Specifications

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

Docket Nos. STN 50-482 and STN 50-483

Ref: SLNRC 84-029, 2/10/84, Same Subject

Dear Mr. Denton:

The purpose of this letter is to forward requested changes to the final draft of the SNUPPS Technical Specifications. Also included as Attachment 1 is a list of all the Technical Specifications presently being discussed with a brief description of the status and action required on each. Any issues on the list forwarded in the referenced letter that are not on the attached list have been satisfactorily closed out. Attachment 2 contains the requested Technical Specification changes.

Very truly yours,

N A Petrick

Nicholas A. Petrick

JHR/dck/1b5

Attachment

cc: J. Neisler/B. Little, USNRC/CAL
W. Schum/K. Whittlesey, USNRC/WC
J. Konklin, USNRC Region III
G. L. Koester, KGE
D. T. McPhee, KCPL
D. F. Schnell, UE

B001
1/1

Attachment 1

SPECIFICATION	SUBJECT	ISSUE	ACTION
Tab. 2.2-1 Tab. 3.3-4	Reactor trip and ESFAS setpoints	Provide final values	The final values for Callaway are attached. Wolf Creek's setpoints will be forwarded when received.
3.1.2.1 3.1.2.5	Boration systems	This specification, applicable in Modes 4, 5, 6, refers to Specification 3.1.2.5 which is applicable in modes 5 and 6 only. The volumes and boron concentration in 3.1.2.5 need to be revised to make the specification applicable in Mode 4 also.	SNUPPS to supply changes to these specifications.
3.2.3	RCS flowrate	The NRC requested that SNUPPS provide background information on how the 2.0% RCS flow uncertainty in this specification was obtained. In a 3/19/84 telecon with the Callaway Project Manager (J. Holonich) SNUPPS reported that the 2% figure was derived using the same methodology as was used for Seabrook, Catawba, and McGuire. The methodology used is a generic one whose results envelope the SNUPPS design.	Callaway PM to inform SNUPPS if further information is required.
3.3.2	ESFAS slave relays	A specification change was requested in SLNRC 84-0?8, 2/27/84. The final decision on several of the slave relays has not been made. In addition, the NRC has requested that the specification list the specific relays that are affected by the 18-month testing.	1. The note to table 4.3-2 has been revised to list the affected relays, see Attach. 2. 2. The SNUPPS Project Managers will determine the status of the decision on relays K615 and K645.
Tab. 3.3-6	Rad Monitoring Instrumentation	Resolve the correct setpoints for ctmt. atmosphere and fuel building exhaust gaseous radioactivity high.	NRC will inform SNUPPS if setpoints provided in SLNRC 84-0029 are acceptable or will revise actions to allow purging.

Attachment 1

SPECIFICATION	SUBJECT	ISSUE	ACTION
Tab. 4.3-4	Seismic Instrumentation	Clarify the ANALOG CHANNEL OPERATIONAL TEST requirements for items 2c, 2d, 2e, and 2f.	Included in Attachment 2
Tab. 3.3-10	Accident Monitoring Instrumentation	Remove instruments that are not presently in the SNUPPS design or which are not approved for operation.	Included in Attachment 2
3.3-4	Turbine Valve Testing	Callaway only - provide a formal transmittal of the final turbine overspeed reliability testing program, and the revised Technical Specification.	Included in Attachment 2
3.4.9.3 3.8.1.2 3.8.2.2 3.8.3.2	RHR Suction Relief Valves	Justify use of the RHR suction relief valves for cold overpressurization protection.	NRC provide final approval of the proposed changes.
Fig. 3.4-3	RCS Cooldown Curves	Provide cooldown curves for Callaway and Wolf Creek.	Included in Attachment 2
3.5.5	Boron Injection Tank	Provide justification for deletion of this specification based on minimum required boron concentration.	Included in Attachment 2
3.6.1.7	Mini-purge	Extend the maximum allowable time for containment purge and increase the allowable leakage rate through the purge valves to .05 La	An appeal meeting was held on 3/15/84. NRC to make a final decision by 3/23/84.
3.6.3.b,c	Containment Isolation Valves	The exclusion of the requirements of specification 3.0.4 was recently deleted by the NRC. This puts the plants in a position where an upward mode change cannot be made if an isolation valve is inoperable, even if the affected penetration is isolated.	SNUPPS Project Manager will assist in reinstating the specification as it was.

Attachment 1

SPECIFICATION	SUBJECT	ISSUE	ACTION
Tab 3.6-1	Containment Isolation Valves	Add two additional values to this table.	Included in Attachment 2.
3.6.4.1	Hydrogen Analyzers	Delete the channel check requirements and change the concentration required for the calibration gas.	NRC complete their review. SNUPPS PM's will follow.
3.8.1.1	Diesel Generator Slow Start Provisions	Revise the specification to require less frequent starts of the diesel generator from ambient conditions in order to minimize diesel wear.	Included in Attachment 2
4.8.1.1.2.d	Diesel Fuel Oil Chemistry Requirements	Revise the testing program for diesel fuel oil to more accurately assess fuel condition.	NRC will add the new program to the Technical Specifications.
Tab 4.8-1	Diesel Generator Testing Requirements	Clarify the note to this table to show that accountability for diesel start failures should begin after all preoperational tests are completed.	Included in Attachment 2
Tab 3.8-1	Containment Penetration Overcurrent Protection Devices	Revise table values to be consistent with the SNUPPS design.	Included in Attachment 2
B 3/4.2.3	Rod Bow Penalty	Provide appropriate values for SNUPPS	Included in Attachment 2
B 3/4.4.5	Steam Generator Inspections	Clarify the types of accidents that require steam generator tube integrity inspections.	Included in Attachment 2
Fig 6.2-1 Fig 6.2-2	Organizational Charts	Provide up-to-date versions of the organizational charts for Callaway.	Callaway will supply these charts.
6.8.1.5.b	NUREG 0737 Procedures	Clarify what provisions of NUREG 0737 have been committed to by SNUPPS.	Included in Attachment 2.

JHR/dck/1b7

Attachment 2

Revised Technical Specifications

and Justifications

CAL ONLY

DRAFT

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*	≤112.3 ~~111.2~~% of RTP*
b. Low Setpoint	8.3	4.56	0	≤25% of RTP*	≤28.3 ~~27.2~~% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	2.4 ~~2.0~~	0.5	0	≤4 ~~5~~% of RTP* with a time constant ≥2 seconds	≤6.3 ~~6.8~~% of RTP* with a time constant ≥2 seconds
4. Power Range, Neutron Flux, High Negative Rate	2.4 ~~2.0~~	0.5	0	≤4 ~~5~~% of RTP* with a time constant ≥2 seconds	≤6.3 ~~6.8~~% of RTP* with a time constant ≥2 seconds
5. Intermediate Range, Neutron Flux	17.0	8.41	0	≤25% of RTP*	≤35.3 ~~31~~% of RTP*
6. Source Range, Neutron Flux	17.0	10.01	0	≤10^5 cps	≤1.6 ~~1.4~~ x 10^5 cps
7. Overtemperature ΔT	6.1 ~~6.7~~	2.76 ~~2.79~~	1.8 ~~0.8~~	See Note 1	See Note 2
8. Overpower ΔT	4.6 ~~4.3~~	1.3	1.2 ~~0.2~~	See Note 3	See Note 4
9. Pressurizer Pressure-Low	5.0 ~~6.8~~	2.21 ~~0.71~~	2.0 ~~1.5~~	≥1885 ~~1900~~ psig	≥1874 ~~1886~~ psig
10. Pressurizer Pressure-High	7.5 ~~3.1~~	4.96 ~~0.71~~	1.0 ~~1.5~~	≤2385 psig	≤2400 ~~2396~~ psig
11. Pressurizer Water Level-High	8.0 ~~5.0~~	2.18	2.0 ~~1.5~~	≤92% of instrument span	≤93.8% of instrument span
12. Reactor Coolant Flow-Low	2.5	1.0	1.5	≥90% of loop design flow**	≥89.2% of loop design flow**

*RTP = RATED THERMAL POWER
**Loop design flow = 95,700 gpm

CAL ONLY

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	SENSOR ERROR Z	(S)	TRIP SETPOINT	ALLOWABLE VALUE
13. Steam Generator Water Level Low-Low	33.5 ~~30.0~~	31.18 ~~27.18~~	2.0 ~~1.5~~	33.5 ≥~~32.3~~% of narrow range instrument span	28.0 ≥~~30.4~~% of narrow range instrument span
14. Undervoltage - Reactor Coolant Pumps	7.7 ~~5.0~~	1.33	0	10584 Volts AC ≥~~70% of bus voltage~~	≥10356 Volts AC
15. Underfrequency - Reactor Coolant Pumps	3.3 ~~1.2~~	0	0	≥57.2 Hz	≥57.1 Hz
16. Turbine Trip					
a. Low Fluid Oil Pressure	N.A.	N.A.	N.A.	598.94 ≥~~554.2~~ psig	579.42 ≥~~548.1~~ psig
b. Turbine Stop Valve Closure	N.A.	N.A.	N.A.	≥1% open	≥1% open
17. Safety Injection Input from ESF	N.A.	N.A.	N.A.	N.A.	N.A.

DRAFT

THIS PAGE OPEN PENDING RECEIPT OF INFORMATION FROM THE APPLICANT

CAL ONLY

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	N.A.	N.A.	N.A.	≥1 x 10^{-10} amps	≥6 x 10^{-11} amps
b. Low Power Reactor Trips Block, P-7					
1) P-10 input	N.A.	N.A.	N.A.	≤10% of RTP*	≤~~12.2~~ 12.4% of RTP*
2) P-13 input	N.A.	N.A.	N.A.	<10% of RTP* turbine Impulse pressure equivalent	<~~12.2~~ 12.4% of turbine Impulse pressure equivalent
c. Power Range Neutron Flux, P-8	N.A.	N.A.	N.A.	≤48% of RTP*	≤~~50.2~~ 51.3% of RTP*
d. Power Range Neutron Flux, P-9	N.A.	N.A.	N.A.	≤50% of RTP*	≤~~52.2~~ 53.3% of RTP*
e. Power Range Neutron Flux, P-10	N.A.	N.A.	N.A.	≥10% of RTP*	≥~~7.8~~ 6.7% of RTP*
f. Turbine Impulse Chamber Pressure, P-13	N.A.	N.A.	N.A.	<10% of RTP* Turbine Impulse Pressure Equivalent	<~~12.2~~ 12.4% of RTP* Turbine Impulse Pressure Equivalent
19. Reactor Trip Breakers	N.A.	N.A.	N.A	N.A.	N.A.
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	N.A.

*RTP = RATED THERMAL POWER

DRAFT

CAL ONLY

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where:

ΔT	=	Measured ΔT by RTD Manifold Instrumentation;
$\frac{1 + \tau_1 S}{1 + \tau_2 S}$	=	Lead-lag compensator on measured ΔT;
τ_1, τ_2	=	Time constants utilized in lead-lag compensator for ΔT, τ_1 = 8 s, τ_2 = 3 s;
$\frac{1}{1 + \tau_3 S}$	=	Lag compensator on measured ΔT;
τ_3	=	Time constant utilized in the lag compensator for ΔT, τ_3 = 0 s;
ΔT_0	=	Indicated ΔT at RATED THERMAL POWER;
K_1	=	1.10;
K_2	=	0.0137 ~~0.0138~~/°F;
$\frac{1 + \tau_4 S}{1 + \tau_5 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_4, τ_5	=	Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = 28 s, τ_5 = 4 s;
T	=	Average temperature, °F;
$\frac{1}{1 + \tau_6 S}$	=	Lag compensator on measured T_{avg};
τ_6	=	Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 s;

DRAFT

CAL ONLY

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

T'	$\leq$	588.5°F (Nominal T_{avg} at RATED THERMAL POWER);
K_3	=	0.000671;
P	=	Pressurizer pressure, psig;
P'	=	2235 psig (Nominal RCS operating pressure);
S	=	Laplace transform operator, s^{-1};

and $f_1(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant STARTUP tests such that:

(i) For $q_t - q_b$ between -35% and + 7%, $f_1(\Delta I) = 0$, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(ii) For each percent that the magnitude of $q_t - q_b$ exceeds -35%, the ΔT Trip Setpoint shall be automatically reduced by 1.26% of its value at RATED THERMAL POWER; and

(iii) For each percent that the magnitude of $q_t - q_b$ exceeds +7%, the ΔT Trip Setpoint shall be automatically reduced by 1.05% of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.8 ~~(3.0)~~% of ΔT span.

DRAFT

THIS PAGE OPEN PENDING RECEIPT OF
INFORMATION FROM THE APPLICANT

CAL ONLY

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \frac{(1 + \tau_1 S)}{(1 + \tau_2 S)} \left(\frac{1}{1 + \tau_3 S}\right) \le \Delta T_o \{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right) \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 [T \left(\frac{1}{1 + \tau_6 S}\right) - T''] - f_2(\Delta I)\}$$

Where: ΔT = Measured ΔT by RTD Manifold Instrumentation;

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead-lag compensator on measured ΔT;

τ_1, τ_2 = Time constants utilized in lead-lag compensator for ΔT, τ_1 = 8 s., τ_2 = 3 s;

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT;

τ_3 = Time constant utilized in the lag compensator for ΔT, τ_3 = 0 s;

ΔT_o = Indicated ΔT at RATED THERMAL POWER;

K_4 = 1.085 ~~1.09~~;

K_5 = 0.02/°F for increasing average temperature and 0 for decreasing average temperature;

$\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the rate-lag compensator for T_{avg} dynamic compensation;

τ_7 = Time constant utilized in the rate-lag compensator for T_{avg}, τ_7 = 10 s;

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg};

τ_6 = Time constant utilized in the measured T_{avg} lag compensator, τ_6 = 0 s;

CAL ONLY

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6	=	0.00128/°F for $T > T''$ and $K_6 = 0$ for $T \leq T''$;
T	=	Average Temperature, °F;
T''	=	Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, ≤ 588.5°F);
S	=	Laplace transform operator, s^{-1}; and
$f_2(\Delta I)$	=	0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than ~~3.0%~~ 2.8 of ΔT span.

DRAFT

Cal Only

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Safety Injection (Reactor Trip, Phase "A" Isolation, Feedwater Isolation, Turbine Trip, Component Cooling Water, Auxiliary Feedwater - Motor - Driven Pump, Emergency Diesel Generator Operation, Containment Cooling, and Essential Service Water Operation)					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure - High-1	3.6 ~~2.5~~	0.71	2.0 ~~1.5~~	≤ 3.5 psig	4.6 ≤ ~~4.0~~ psig
d. Pressurizer Pressure - Low	18.6 ~~13.0~~	14.91 ~~10.71~~	2.0 ~~1.5~~	≥ 1849 psig	1834 ≥ ~~1839~~ psig
e. Steam Line Pressure - Low	19.6 ~~14.2~~	14.81 ~~10.71~~	2.0 ~~1.5~~	615 ≥ ~~585~~ psig	571 ≥ ~~564~~ psig*
2. Containment Spray					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	N.A.

DRAFT

CAL ONLY

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
2. Containment Spray (Continued)					
c. Containment Pressure-High-3	4.3 ~~5.0~~	0.71	2.0 ~~1.5~~	$\leq$ 27.0 psig	$\leq$ 28.3 ~~28.0~~ psig
3. Containment Isolation					
a. Phase "A" Isolation					
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
b. Phase "B" Isolation					
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	N.A.
3) Containment Pressure-High-3	4.3 ~~5.0~~	0.71	2.0 ~~1.5~~	$\leq$ 27.0 psig	$\leq$ 28.3 ~~28.0~~ psig
c. Containment Purge Isolation					
1) Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	N.A.

DRAFT

CAL ONLY

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
3. Containment Isolation (Continued)					
3) Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	N.A.	N.A.	N.A.	N.A.	N.A.
4) Phase "A" Isolation	See Item 3.a. above for all Phase "A" Isolation Trip Setpoints and Allowable Values.				
4. Steam Line Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	N.A.
c. Containment Pressure-High-2	4.3 ~~5.0~~	0.71	2.0 ~~1.5~~	≤ 17.0 psig	≤ 18.3 ~~18.0~~ psig
d. Steam Line Pressure-Low	19.6 ~~14.2~~	14.81 ~~10.71~~	2.0 ~~1.5~~	≥ 615 ~~585~~ psig	≥ 571 ~~564~~ psig*
e. Steam Line Pressure Negative Rate - High	3.0 ~~8.0~~	0.5	0	≤ -100 psi/s	≤ -124 ~~111.6~~ psi/s**
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A	N.A.	N.A.	N.A.

DRAFT

CAL ONLY

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
5. Feedwater Isolation (Continued)					
b. Steam Generator Water Level-High-High	5	2.18	2.0 ~~1.5~~	$\leq$ 78% of narrow range instrument span	$\leq$ 79.8% of narrow range instrument span
c. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
6. Auxiliary Feedwater					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	N.A.
c. Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	N.A.	N.A.	N.A.	N.A.	N.A.
d. Steam Generator Water Level-Low-Low					
1) Start Motor-Driven Pumps	23.5 ~~30.0~~	21.18 ~~27.18~~	2.0 ~~1.5~~	$\geq$ 23.5 ~~32.2~~% of narrow range instrument span	$\geq$ 22 ~~30.4~~% of narrow range instrument span
2) Start Turbine-Driven Pump	23.5 ~~30.0~~	21.18 ~~27.18~~	2.0 ~~1.5~~	$\geq$ 23.5 ~~32.2~~% of narrow range instrument span	$\geq$ 22 ~~30.4~~% of narrow range instrument span

DRAFT

CAL ONLY

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
6. Auxiliary Feedwater (Continued)					
e. Safety Injection-Start Motor-Driven Pumps	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
f. Loss-of-Offsite Power-Start Turbine-Driven Pump	N.A.	N.A.	N.A.	N.A.	N.A.
g. Trip of All Main Feedwater Pumps-Start Motor-Driven Pumps	N.A.	N.A.	N.A.	N.A.	N.A.
h. Auxiliary Feedwater Pump Suction Pressure-Low (Transfer to ESW)	N.A.	N.A.	N.A.	23.40 ~~≥22.03~~	21.72 ~~≥21.89~~
7. Automatic Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	N.A.
b. RWST Level-Low-Low Coincident with Safety Injection	3.4 ~~N.A.~~	1.21 ~~N.A.~~	2.0 ~~N.A.~~	≥ 36% ~~(≥18%)~~	≥ 35.2 ~~(≥15%)~~
	See Item 1. above for Safety Injection Trip Setpoints and Allowable Values.				
8. Loss of Power					
a. 4 kV Undervoltage -Loss of Voltage	N.A.	N.A.	N.A.	83V (120V Bus) w/1s delay	83+0,-8.3V (120V Bus) w/1+0.2,-0.5s delay

THIS PAGE OPEN PENDING RECEIPT OF INFORMATION FROM THE APPLICANT

DRAFT

CAL ONLY

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. Loss of Power (Continued)					
b. 4 kV Undervoltage -Grid Degraded Voltage	N.A.	N.A.	N.A.	104.5V (120V Bus) w/119s delay	104.5+2.6,-0V (120V Bus) w/119 ± 11.6s delay
9. Control Room Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	N.A.
c. Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	N.A.	N.A.	N.A.	N.A.	N.A.
d. Phase "A" Isolation	See Item 3.a. above for all Phase "A" Isolation Trip Setpoints and Allowable Values.				
10. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	≤ 1970 psig	≤ 1981 psig
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	N.A.

DRAFT

CAL ONLY

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

DRAFT

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection (Reactor Trip, Phase "A" Isolation, Feedwater Isolation, Turbine Trip, Component Cooling Water, Auxiliary Feedwater-Motor-Driven Pump, Emergency Diesel Generator Operation, Containment Cooling, and Essential Service Water Operation)								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(3)	1, 2, 3, 4
c. Containment Pressure-High-1	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. Containment Spray								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(3)	1, 2, 3, 4
c. Containment Pressure-High-3	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

CAL ONLY

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays (SSPS)	N.A	N.A.	N.A.	N.A.	M(1)	M(1)	Q(3)	1, 2, 3, 4
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
b. Phase "B" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Containment Pressure-High-3	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Containment Purge Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(3)	1, 2, 3, 4
3) Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	N.A.	N.A.	N.A.	N.A.	M(2)	N.A.	N.A.	1, 2, 3, 4
4) Phase "A" Isolation	See Item 3.a. above for all Phase "A" Isolation Surveillance Requirements.							

DRAFT

CAL ONLY

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Steam Line Isolation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Containment Pressure-High-2	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Line Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure-Negative Rate-High	S	R	M	N.A.	N.A.	N.A.	N.A.	3, 4
5. Turbine Trip and Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relay	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(3)	1, 2
b. Steam Generator Water Level-High-High	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2
c. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
6. Auxiliary Feedwater								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3

DRAFT

CAL ONLY

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
6. Auxiliary Feedwater (Continued)								
c. Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	N.A.	N.A.	N.A.	N.A.	M(2)	N.A.	N.A.	1, 2, 3
d. Steam Generator Water Level-Low-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
f. Loss-of-Offsite Power	N.A.	R	N.A.	M	N.A.	N.A.	N.A.	1, 2, 3
g. Trip of All Main Feedwater Pumps	N.A.	N.A.	N.A	R	N.A.	N.A.	N.A.	1, 2
h. Auxiliary Feedwater Pump Suction Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
7. Automatic Switchover to Containment Sump								
a. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(3)	1, 2, 3, 4
b. RWST Level - Low-Low Coincident With Safety Injection	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
	See Item 1. above for all Safety Injection Surveillance Requirements.							
8. Loss of Power								
a. 4 kV Undervoltage-Loss of Voltage	N.A.	R	N.A.	M	N.A.	N.A.	N.A.	1, 2, 3, 4

DRAFT

CAL ONLY

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
8. Loss of Power (Continued)								
b. 4 kV Undervoltage-Grid Degraded Voltage	N.A.	R	N.A.	M	N.A.	N.A.	N.A.	1, 2, 3, 4
9. Control Room Isolation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	All
b. Automatic Actuation Logic and Actuation Relays (SSPS)	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q(3)	All
c. Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	N.A.	N.A.	N.A.	N.A.	M(2)	N.A.	N.A.	All
d. Phase "A" Isolation	See Item 3.a. above for all Phase "A" Isolation Surveillance Requirements.							
10. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure, P-11	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3

DRAFT

TABLE NOTATION

(1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.
(2) Continuity check may be excluded from the ACTUATION LOGIC TEST.
(3) See INSERT on following page

INSERT

(3) Except Relays K602, K615, K622, K630, K645, K740 and K741 which shall be tested, as a minimum, once per 18 months during refueling and each shutdown to COLD SHUTDOWN unless they have been tested within the previous 90 days.

COPPER CONTENT : CONSERVATIVELY ASSUMED TO BE 0.10 WT%

RT_{NDT} INITIAL : 40°F

RT_{NDT} AFTER 16 EFPY : 1/4T, 110°F
3/4T, 87°F

CURVE APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 16 EFPY AND CONTAINS MARGINS OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS

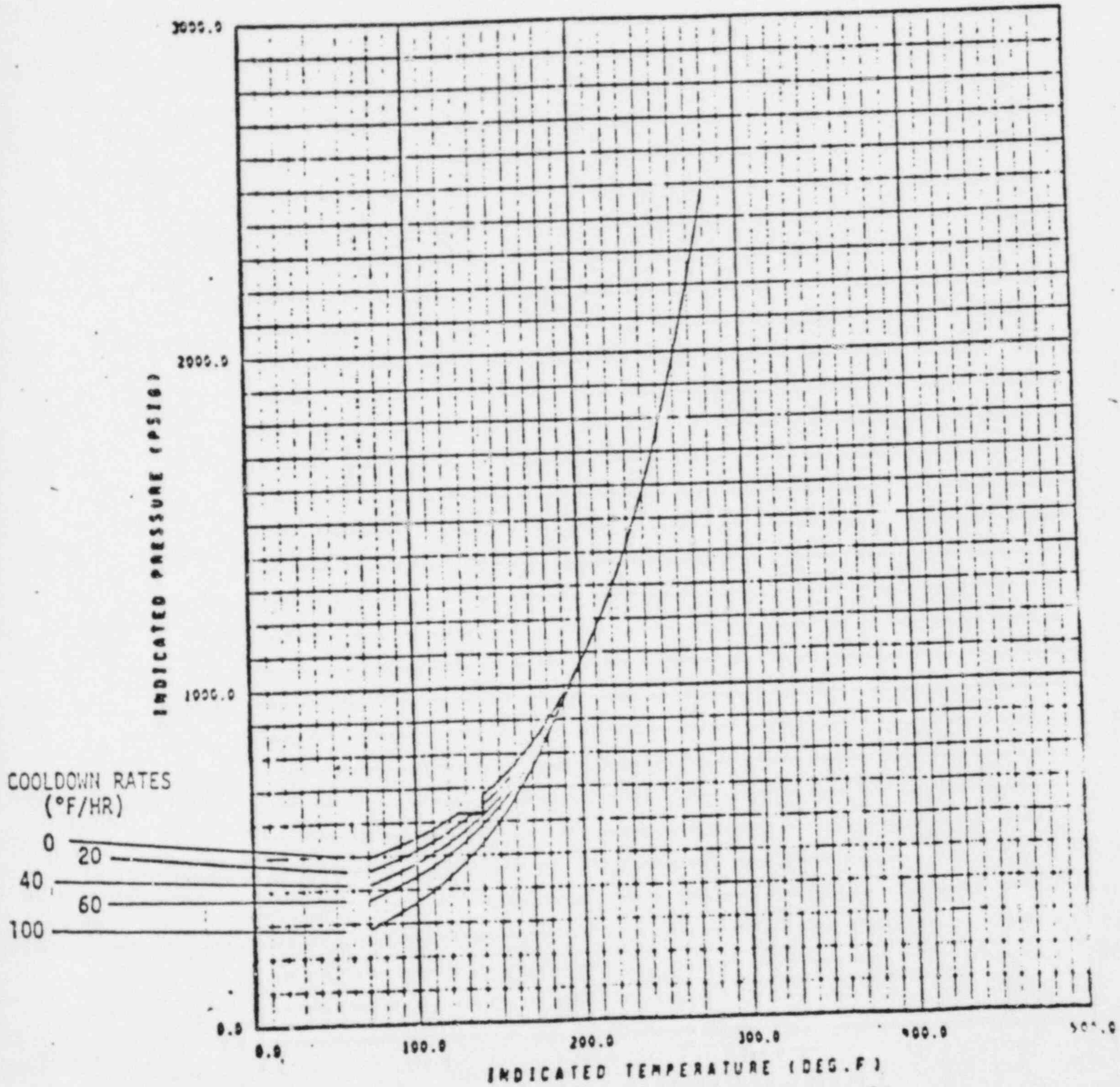

Figure 2 Wolf Creek Reactor Coolant System Cooldown Limitations Applicable up to 16 EFPY

COPPER CONTENT : CONSERVATIVELY ASSUMED TO BE 0.10 WT%

RT_{NDT} INITIAL : 50°F

RT_{NDT} AFTER 7 EFPY : 1/4T, 110°F
3/4T, 87°F

CURVE APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 7 EFPY AND CONTAINS MARGINS OF 10°F AND 60 PSIG FOR POSSIBLE INSTRUMENT ERRORS

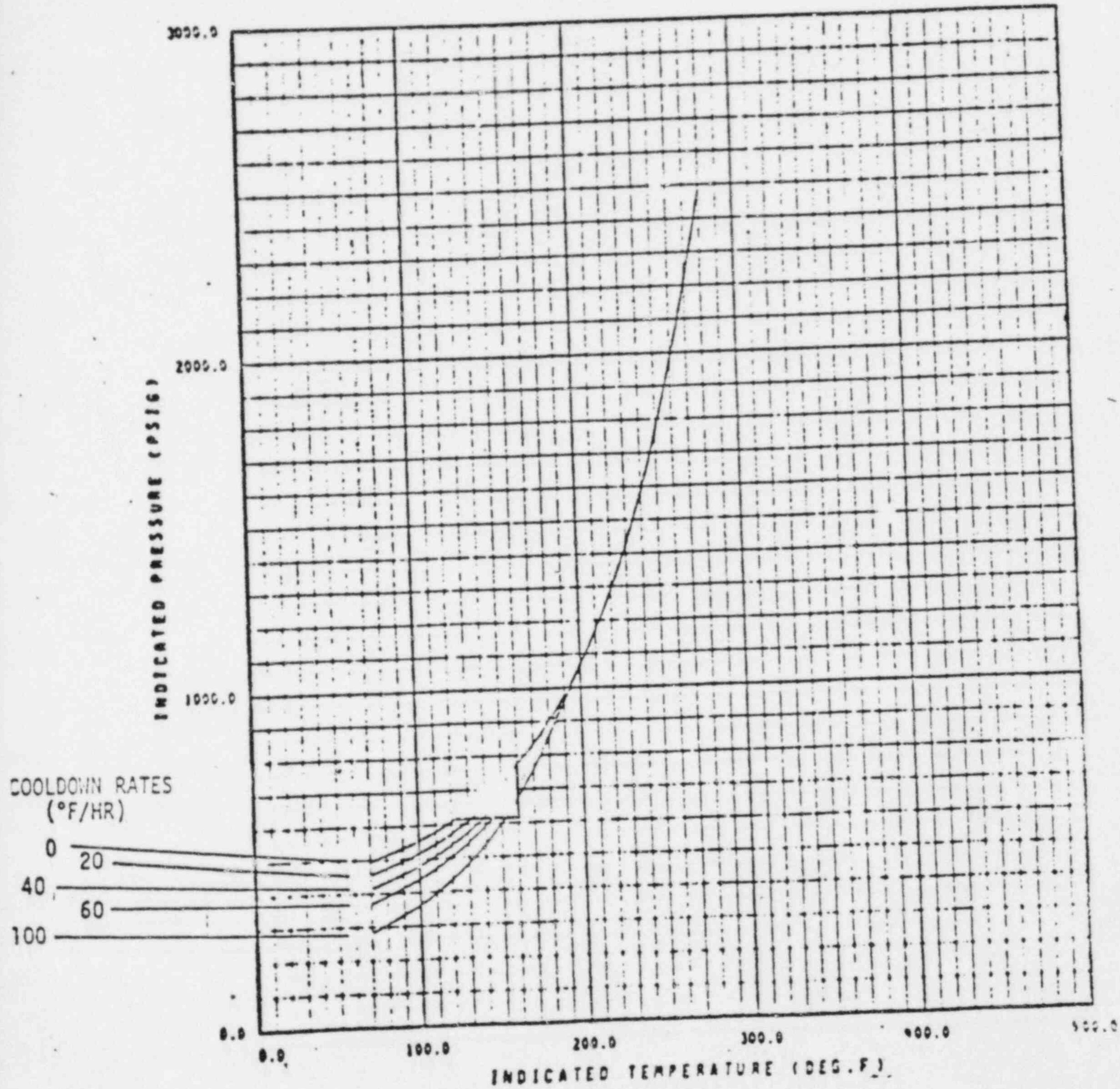

Figure 4 Callaway Unit 1 Reactor Coolant System Cooldown Limitations Applicable up to 7 EFPY

DRAFT

TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST
1. Triaxial Peak Recording Accelerographs			
a. Radwaste Base Slab	N.A.	R	N.A.
b. Control Room	N.A.	R	N.A.
c. ESW Pump Facility	N.A.	R	N.A.
d. Ctmt Structure	N.A.	R	N.A.
e. Auxiliary Bldg. SI Pump Suction	N.A.	R	N.A.
f. SGB Piping	N.A.	R	N.A.
g. SGB Support	N.A.	R	N.A.
2. Triaxial Time History and Response Spectrum Recording System, Monitoring the Following Accelerometers (Active)			
a. Ctmt. Base Slab	M	R	SA
b. Ctmt. Oper. Floor	M	R	SA
c. Reactor Support	M	R	SA**
d. Aux. Bldg. Base Slab	M	R	SA**
e. Aux. Bldg. Control Room Air Filters	M	R	SA**
f. Free Field	M	R	SA**
3. Triaxial Response-Spectrum Recorder (Passive)			
Ctmt. Base Slab	N.A.	R	N.A.*
4. Triaxial Seismic Switches			
a. OBE Ctmt. Base Slab	M	R	SA
b. SSE Ctmt. Base Slab	M	R	SA
c. OBE Ctmt. Oper. Fl.	M	R	SA
d. SSE Ctmt. Oper. Fl.	M	R	SA
e. System Trigger	M	R	SA

*Checking at the Main Control Board Annunciators for contact closure output in the Control Room shall be performed at least once per 184 days.

** The bistable trip setpoint is not determined during the performance of this ANALOG CHANNEL OPERATIONAL TEST.

Justification - Table 4.3-4

The double asterisked note was added to reflect the SNUPPS design. Existing test facilities do not allow for the determination of the bistable trip setpoint (as required by the definition of ANALOG CHANNEL OPERATIONAL TEST). These test facilities do, however, allow a functional test to be performed which will verify the operability of these instruments.

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Containment Pressure		
a. Normal Range	2	1
b. Extended Range	2	1
2. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	2	1
3. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	2	1
4. Reactor Coolant Pressure - Wide Range	2	1
5. Pressurizer Water Level	2	1
6. Steam Line Pressure	2/steam generator	1/steam generator
7. Steam Generator Water Level - Narrow Range	1/steam generator	1/steam generator
8. Steam Generator Water Level - Wide Range	1/steam generator	1/steam generator
9. Refueling Water Storage Tank Water Level	2	1
10. Containment Hydrogen Concentration Level	2	1
11. Auxiliary Feedwater Flow Rate	1/steam generator	1/steam generator
~~12. Reactor Coolant System Subcooling Margin Monitor~~	~~2~~	~~1~~
13. PORV Position Indicator*	1/Valve	1/Valve
14. PORV Block Valve Position Indicator**	1/Valve	1/Valve
15. Safety Valve Position Indicator	1/Valve	1/Valve
16. Containment Water Level	2	1
17. Containment Radiation Level (High Range)	2	1
18. Thermocouple/Core Cooling Detection System	4/core quadrant	2/core quadrant

*Not applicable if the associated block valve is in the closed position.

**Not applicable if the block valve is verified in the closed position and power is removed.

DRAFT

DRAFT

TABLE 3.3-10 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
~~19. Reactor Coolant Radiation Level~~	~~1/loop~~	~~1/loop*~~
20. Unit Vent - High Range Noble Gas Monitor	1/stack	1/stack~~*~~

~~*If inoperable, Action a. in Specification 3.3.3.6 shall be applicable~~.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure	M	R
2. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	M	R
3. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	M	R
4. Reactor Coolant Pressure - Wide Range	M	R
5. Pressurizer Water Level	M	R
6. Steam Line Pressure	M	R
7. Steam Generator Water Level - Narrow Range	M	R
8. Steam Generator Water Level - Wide Range	M	R
9. Refueling Water Storage Tank Water Level	M	R
10. Containment Hydrogen Concentration Level	M	R
11. Auxiliary Feedwater Flow Rate	M	R
~~12. Reactor Coolant System Subcooling Margin Monitor~~	~~M~~	~~R~~
13. PORV Position Indicator*	M	N.A.
14. PORV Block Valve Position Indicator**	M	N.A.
15. Safety Valve Position Indicator	M	N.A.
16. Containment Water Level	M	R
17. Containment Radiation Level (High Range)	M	R***
18. Thermocouple/Core Cooling Detection System	M	R

*Not applicable if the associated block valve is in the closed position.

**Not applicable if the block valve is verified in the closed position and power is removed.

***CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for range decades above 10R/h and a one point calibration check of the detector below 10R/h with an installed or portable gamma source.

DRAFT

TABLE 4.3-7 (Continued)

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
~~19. Reactor Coolant Radiation Level~~	~~M~~	~~R~~
20. Unit Vent - High Range Noble Gas Monitor	M	R

Justification - Tables 3.3-10 and 4.3-7

The Reactor Coolant Radiation Level Monitor has been deleted from these tables since the SNUPPS design does not incorporate the instrument. Further, no qualified instrument is commercially available at this time.

The Reactor Coolant System Subcooling Margin Monitor has also been removed from these tables. This was done because SNUPPS, in response to Generic Letter 82-33, (transmitted by SLNRC 83-0019, dated April 15, 1983) indicated that this instrument would not be operable until startup after the first refueling outage at Callaway and at Wolf Creek. This schedule has been approved by the NRC staff. In further discussions with the NRC, SNUPPS was informed that other instrumentation designed to the guidance of Regulatory Guide 1.97, Rev. 2, was to be included in these tables. These are as follows:

1. Qualified Source Range Instrumentation
2. Reactor Vessel Water Level Instrumentation System
3. Radiation Monitors for Releases from Steam Generator Safety/ Relief Valves or Atmospheric Dump Valves
4. Auxiliary Feedwater Pump Turbine Exhaust Monitor

In SLNRC 83-0019, dated April 15, 1983, SNUPPS indicated that these instruments would also not be operable until startup after the first refueling at Callaway and at Wolf Creek. This schedule has been approved by the NRC staff.

In view of the fact that the NRC review of the SNUPPS design comparison to Regulatory Guide 1.97, Rev. 2, has yet to be completed and documented in a Safety Evaluation Report, it is inappropriate for these items to be reflected in Technical Specifications. Technical Specifications should be a reflection of plant design and should not be used as a tool for implementing design or schedular changes. SNUPPS contends that the NRC should complete and document their review of the SNUPPS design comparison to Regulatory Guide 1.97, Rev. 2, prior to deciding to include any of these instruments in Technical Specifications.

CALLAWAY SPECIFIC

PROOF & REVIEW COPY

3/4.3.4 TURBINE OVERSPEED PROTECTION

LIMITING CONDITION FOR OPERATION

3.3.4 At least one Turbine Overspeed Protection System shall be OPERABLE.

APPLICABILITY: MODES 1, 2,* and 3.*

ACTION:

a. With one stop valve or one governor valve per high pressure turbine steam line inoperable and/or with one reheat stop valve or one reheat intercept valve per low pressure turbine steam line inoperable, restore the inoperable valve(s) to OPERABLE status within 72 hours, or close at least one valve in the affected steam lines, or isolate the turbine from the steam supply within the next 6 hours.

b. With the above required Turbine Overspeed Protection System otherwise inoperable, within 6 hours isolate the turbine from the steam supply.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.4.1 The provisions of Specification 4.0.4 are not applicable.

4.3.4.2 ~~The above required Turbine Overspeed Protection System shall be demonstrated OPERABLE:~~ SEE INSERT FOR SPECIFICATION 4.3.4.2

~~a. At least once per 7 days by cycling each of the following valves through at least one complete cycle from the running position:~~

~~1) Four high pressure turbine stop valves,~~

~~2) Four high pressure turbine governor valves,~~

~~3) Four low pressure turbine reheat stop valves, and~~

~~4) Four low pressure turbine reheat intercept valves.~~

~~b. At least once per 31 days by direct observation of the movement of each of the above valves thorugh one complete cycle from the running position;~~

~~c. At least once per 18 months by performance of a CHANNEL CALIBRATION on the Turbine Overspeed Protection Systems; and~~

~~d. At least once per 40 months by disassembling at least one of each of the above valves and performing a visual and surface inspection of valve seats, disks and stems and verifying no unacceptable flaws or corrosion.~~

* Specification not applicable with all main steam isolation valves and associated bypass valves in the closed position and all other steam flow paths to the turbine isolated.

Insert for Specification 4.3.4.2

4.3.4.2 The Turbine Overspeed Protection System will be maintained, calibrated, tested and inspected in accordance with the Callaway Plant "Turbine Overspeed Protection Reliability Program." Adherence to this program will demonstrate Operability. The program and any subsequent revisions will be reviewed and approved in accordance with Callaway Plant Technical Specification section 6.0. Revisions will be made in accordance with the provisions of 10CFR50.59.

RESPONSIBILITIES (Continued)

m. Review of Unit operations to detect potential hazards to nuclear safety; ~~and~~

n. Investigations or analysis of special subjects as requested by the Chairman of the NSRB; and

o. Review of Callaway Plant TURBINE OVERSPEED PROTECTION RELIABILITY PROGRAM and revisions thereto.

6.5.1.7 The ORC shall:

a. Recommend in writing to the Manager, Callaway Plant approval or disapproval of items considered under Specifications 6.5.1.6a. through e., i., j., k., l. and O., above;

b. Render determinations in writing with regard to whether or not each item considered under Specifications 6.5.1.6b. through e., and m., above, constitutes an unreviewed safety question; and

c. Provide written notification within 24 hours to the Vice President-Nuclear and the Nuclear Safety Review Board of disagreement between the ORC and the Manager, Callaway Plant; however, the Manager, Callaway Plant shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1 above.

RECORDS

6.5.1.8 The ORC shall maintain written minutes of each ORC meeting that, at a minimum, document the results of all ORC activities performed under the responsibility and authority provisions of these Technical Specifications. Copies shall be provided to the Vice President-Nuclear and the Nuclear Safety Review Board.

6.5.2 NUCLEAR SAFETY REVIEW BOARD (NSRB)

FUNCTION

6.5.2.1 The NSRB shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations,

b. Nuclear engineering,

c. Chemistry and radiochemistry,

d. Metallurgy,

e. Instrumentation and control,

f. Radiological safety,

g. Mechanical and electrical engineering, and

h. Quality assurance practices.

The NSRB shall report to and advise the Vice President-Nuclear on those areas of responsibility stated in Specifications 6.5.2.8 and 6.5.2.9.

SAFETY LIMIT VIOLATION (Continued)

c. The Safety Limit Violation Report shall be submitted to the Commission, the NSRB and the Vice President-Nuclear within 14 days of the violation; and

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A, of Regulatory Guide 1.33, Revision 2, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Section 7.1 of ~~in~~ Generic Letter 82-33;

c. Plant Security Plan implementation;

d. Radiological Emergency Response Plan implementation;

e. PROCESS CONTROL PROGRAM implementation,

f. OFFSITE DOSE CALCULATION MANUAL implementation, ~~and~~

g. Quality Assurance Program for effluent and environmental monitoring, and

h. Turbine Overspeed Protection Reliability Program

6.8.2 Each procedure and administrative policy of Specification 6.8.1 above, and changes thereto, including temporary changes shall be reviewed prior to implementation as set forth in Specification 6.5 above.

6.8.3 The plant Administrative Procedures and changes thereto shall be reviewed in accordance with Specification 6.5.1.6 and approved in accordance with Specification 6.5.3.1. The associated implementing procedures and changes thereto shall be reviewed and approved in accordance with Specification 6.5.3.1.

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the recirculation portion of the Containment Spray System, Safety Injection System, Chemical and Volume Control System, and RHR System. The program shall include the following:

1) Preventive maintenance and periodic visual inspection requirements, and

PROCEDURES AND PROGRAMS (Continued)

2) Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1) Training of personnel,

2) Procedures for monitoring, and

3) Provisions for maintenance of sampling and analysis equipment.

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

1) Identification of a sampling schedule for the critical variables and control points for these variables,

2) Identification of the procedures used to measure the values of the critical variables,

3) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

4) Procedures for the recording and management of data,

5) Procedures defining corrective action for all off-control point chemistry conditions, and

6) A procedure identifying: (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Post-accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

1) Training of personnel,

2) Procedures for sampling and analysis, and

3) Provisions for maintenance of sampling and analysis equipment.

e. Turbine Overspeed Protection Reliability Program

 (see following page)

e. Turbine Overspeed Protection Reliability Program

A program to increase the assurance that the turbine overspeed protection system functions, if challenged, and to assure structural integrity of turbine components which could result in missile generation in the event of an actual overspeed occurrence. The program shall include the following:

1) Periodic testing and inspection requirements,
2) Specification of test and inspection intervals, and
3) Administrative restrictions and procedural guidance for program implementation such as: record keeping; reporting, evaluation and disposition of discrepancies; review and approval of revisions to the program; and authorization(s) required to deviate from the program guidelines.

SAFETY EVALUATION

The turbine-generator placement and orientation at Callaway Unit 1 is favorable relative to essential plant buildings; that is, there are no structures, systems, or components important to safety inside the low trajectory missile (LTM) strike zone. The NRC staff has concluded that systems important to safety at Callaway Unit 1 are adequately protected from low-trajectory turbine missiles. The NRC staff has also reviewed the high-trajectory turbine missile (HTM) analysis and concluded that the probability of Callaway systems important to safety being struck by HTM's is sufficiently low that the risk rate for unacceptable damage due to HTM's is less than 10-7 per year. The NRC staff considers this an acceptable risk rate for the loss of an essential system from a single event.

The turbine-generator for Callaway Unit 1 was manufactured by General Electric Company. The turbine (No. 170X732) is a 14 stage tandem-compound unit consisting of a double flow high-pressure section and three double flow low-pressure sections rated at 1800 RPM and 1192498KW. The generator (No. 180X732) is a 4 pole 60 HZ machine with a gross output of 1186 MWe at a nominal plant exhaust pressure of 2.61 in. mercury (absolute). The turbine-generator is equipped with an electrohydraulic control (EHC) system which provides for steam pressure control, speed control, load control, and steam control valve positioning. In order to produce missiles, significant overspeeding (beyond design speed of 128%) of the turbine must occur. The EHC is a highly reliable system, employing three electrical and one mechanical speed inputs. Logic signals are processed in both electronic and hydraulic channels for redundancy. Valve opening actuation is provided by a 1,600-psig hydraulic system which is totally independent of the bearing lubrication system. Valve closing actuation is provided by springs and aided by steam forces following the reduction or relief of hydraulic fluid pressure which leads to valve closing and consequent shutdown. The main steam turbine inlet valves, consisting of 4 single disc type Main Stop Valves and 4 venturi seat poppet-type Control Valves, are provided in series arrangements: the stop valves actuated by either of two overspeed trip signals, followed by the control valves modulated by the speed-governing system, and tripped by either overspeed-trip signal. The 6 Combined Intermediate valves are arranged in series-pairs, with an intermediate stop valve and intercept valve in one casing. The closure of either one of the two valves will close off the corresponding steam line. Thus, a single failure at the instant of load loss would be required, involving combinations of undetected electronic faults and/or mechanically stuck valves and/or hydraulic fluid contamination. The probability of such joint occurrences is extremely low, due both to the inherently high reliability of the design of the components and frequent inservice inspection. Also included in this system are 4 air operated check type Extraction Nonreturn Valves used to prevent stored energy in the extractions lines from overspeeding the turbine. There have been no runaways of General Electric turbines equipped with EHC.

Although the turbine control and overspeed protection system is not relied on to perform a safety function, it controls a plant process that has potential to impact plant safety. The objective of the valve testing is to assure high valve operability and reliability in order to minimize the probability of generating destructive missiles that could damage safety related equipment and thereby prevent safe shutdown of the plant. The turbine control and overspeed protection system is designed to control turbine action under all normal and abnormal conditions to assure that a turbine trip from full load will not cause the turbine to overspeed beyond acceptable limits, thus minimizing the

probability of generating turbine missiles. Nuclear turbine valves have proven to be extremely reliable in service as evidenced by the lack of failures over the many years of nuclear plant operation. This proven reliability can also be attributed to the all volatile chemical treatment of feedwater which minimizes steam generator carry over and essentially eliminates valve failure due to scale buildup on the valve moving parts.

The inspection, testing, maintenance and calibration requirements have been reviewed by General Electric and found acceptable.

Union Electric Company requests the proposed change to the technical specifications to allow required flexibility and unit reliability during peak load demands or other operational limitations which is not currently permitted by Standard Technical Specifications. The proposed program facilitates the maintenance of a high degree of reliability of the Turbine Overspeed Protection System, and allows the required operational flexibility. The proposed specifications and attendant program provide for deferral of testing when, in the opinion of Union Electric Company Management, such testing would: (a) result in load reduction to facilitate testing during periods when the reduction could not be made without jeopardizing system reliability; or (b) the increased probability of a forced outage resulting from the testing poses an unacceptable risk to the reliability of the system. The Main Stop and Combined Intercept Valve testing will result in power pertubations on the order of ±2% of full power, and the testing of the Control Valves will require load reductions on the order of 10-15% of full power. The valve testing per se requires a relative short time interval. However, since the Callaway reactor core is operated under constant axial offset limits to preclude Xenon oscillations, the load changes to accommodate such testing would require several hours, with attendant boration and dilution of the reactor coolant system. Since Callaway Unit 1 will be a base load unit, it may be necessary during peak load conditions to defer the testing to avoid subjecting the system to this load loss, transient and further instability. The decision to defer testing as described in section 5.1.4 of the TOPRP will be based on Union Electric Company Management's evaluation of the need based on the above criteria.

The Union Electric Company On-Site Review Committee has reviewed this proposed technical specification change and has determined that this change does not involve an unreviewed safety question. Subsequent changes to the program as presently described in the TOPRP document in scope and/or schedule resulting from on-going review by Union Electric Company, operating experience at Callaway Unit 1, operating experience with similar General Electric units and results of studies will be reviewed and approved as stated in the revised plant technical specifications section 6.0 and implemented in accordance with provisions of 10 CFR 50.59.

In summary, the basis and reasons for requesting the proposed change in the Callaway Unit 1 Plant Technical Specifications are:

1. The Callaway Unit 1 turbine-generator is favorably oriented thus minimizing the probability of low and high trajectory missile strikes to safety related equipment.

2. Nuclear turbine valves and protection systems have proven to be extremely reliable in service as evidenced by the lack of failures over the many years of nuclear plant operation.

3. The proposed Union Electric TOPRP exceeds the NRC staff requirements in that Main Stop and Combined Intercept Valves will be tested on a daily vice weekly basis and visual confirmation of valve stem movement on a weekly vice monthly basis. The inspection, testing, maintenance and calibration requirements have been reviewed by General Electric and found acceptable.

4. The Standard Technical Specifications do not allow the operational flexibility needed under abnormal grid system operation or abnormal plant operation. Testing of turbine valves under such conditions could cause unacceptable loss of generation capability, system instability, transients, or loss of large portions of the grid system.

5. The proposed revision to Standard Technical Specifications 3/4.3.4 "Turbine Overspeed Protection" does not involve a significant hazards consideration.

6. The proposed Union Electric TOPRP coupled with installed turbine-generator protective features and an inplace inspection program of the low pressure turbine discs provides reasonable assurance of a low overspeed missile generation probability.

7. All changes to the TOPRP in scope and/or schedule resulting from on-going review by Union Electric, operating experience at Callaway Unit 1, operating experience with other similar General Electric units and results of studies will be reviewed and approved as stated in the revised plant technical specifications section 6.0 and implemented in accordance with the provisions of 10 CFR 50.59.

5.2.2 The calibration program for the turbine overspeed protection program shall include the following at least once per 18 months or following major maintenance on the turbine generator or the overspeed protection system.

5.2.2.1 Mechanical Overspeed Trip Calibration - the mechanical overspeed trip test is designed to verify calibration of the turbine mechanical overspeed trip system. The turbine speed is increased to the trip setpoint of 109.5% and speed at which the trip occurs is recorded. If the as-found trip value is out of tolerance, the trip setpoint is adjusted and the test is repeated.

5.2.2.2 Backup Overspeed Trip Calibration - the BOST is designed to verify calibration of the electrical Auxiliary Speed Sensor unit. In the NORMAL mode this trip is set at 110% of rated speed and is a backup to the mechanical overspeed trip. In the STANDBY mode this trip is reduced to 105% and provides the first line of protection. The actual speed at which the trip occurs is compared to the trip setpoint. If the as-found trip value is out of tolerance, the trip setpoint is adjusted and the test is repeated.

5.2.3 Maintenance and Inspection Program.

5.2.3.1 The inservice inspection examination of valves important to overspeed protection shall include the following:

5.2.3.1.1 All Main Stop, Main Control, Combined Intercept and Extraction Nonreturn valves will be inspected once during the first three years. Inspection of all valves of one type will be conducted if any unusual condition is discovered. Subsequent inspections will be scheduled so that each valve is inspected at 3- to 5-year intervals. The inspections will be conducted for:

a. Wear of linkages and stem packings

b. Erosion of valve seats and stems

c. Deposits on stems and other valve parts which could interfere with valve operation

d. Distortions, misalignments

5.2.3.2 Tightness tests of the main stop and control valves are performed at least once per 18 months by checking the coastdown characteristics of the turbine from no load with each set of four valves closed alternately. Platforms provided for valve maintenance permit observation of the valve motion.

5.2.3.3 The inservice inspection program for the turbine assembly includes the disassembly of the turbine and complete inspection of all normally inaccessible parts, such as couplings, coupling bolts, turbine shafts low-pressure turbine buckets, low-pressure wheels, and high-pressure rotors. During plant shutdown coinciding with the inservice inspection schedule for ASME Section III components, as required by the ASME Boiler and Pressure Vessel Code, Section XI, turbine inspection is done in sections during the refueling outages so that in 10 years total inspection has been completed at least once.

5.2.3.3.1 This inspection consists of visual and surface examinations as indicated below:

a. Visual examination of all accessible surfaces of rotors and wheels

b. Visual and surface examination of all low-pressure buckets

c. 100-percent visual examination of couplings and coupling bolts

5.2.3.3.2 Inservice inspection of the bores and keyways of the low-pressure turbine discs will be in accordance with the manufacturer's recommendations.

6.0 REPORTING

6.1 The Main Stop Control, Combined Intercept, and Extraction Nonreturn valves shall be included in the Nuclear Plant Reliability Data System (NPRDS). Deficiencies shall be reported and included in the data bank, and reviewed so that appropriate changes may be made in the Callaway Plant program based on reliability information.

7.0 RECORDS

7.1 Records for the Turbine Overspeed Protection Reliability Program shall be maintained in accordance with the implementing procedures.

8.0 REVISIONS TO THE TURBINE OVERSPEED PROTECTION RELIABILITY PROGRAM

Modifications, deviations, and other changes to the Turbine Overspeed Protection Reliability Program as a result of on-going review by Union Electric Company shall be initiated and processed in accordance with Callaway Plant Administrative Procedures, Callaway Plant Technical Specifications and implemented in accordance with the provisions 10CFR50.59.

9.0 REFERENCES

9.1 Callaway Plant Technical Specifications

9.2 APA-ZZ-00101, "Preparation, Review, Approval, and Control of Plant Procedures"

9.3 APA-ZZ-00140, "Conduct of Engineering and Safety Evaluations"

9.4 Final Safety Analysis Report

9.5 GEK-65907 VOL. I

9.6 GEK-64907 VOL. II

DRAFT

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION		PAGE
EMERGENCY CORE COOLING SYSTEMS (Continued)		
DELETE 3/4.5.5	BORON INJECTION SYSTEM	
	Boron Injection Tank..................................	3/4 5-10 DELETE
3/4.5.~~6~~ 5	REFUELING WATER STORAGE TANK..............................	3/4 5-~~11~~ 10
3/4.6 CONTAINMENT SYSTEMS		
3/4.6.1	PRIMARY CONTAINMENT	
	Containment Integrity..................................	3/4 6-1
	Containment Leakage..................................	3/4 6-2
	Containment Air Locks..................................	3/4 6-4
	Internal Pressure..................................	3/4 6-6
	Air Temperature..................................	3/4 6-7
	Containment Vessel Structural Integrity..................	3/4 6-8
	Containment Ventilation System..........................	3/4 6-11
3/4.6.2	DEPRESSURIZATION AND COOLING SYSTEMS	
	Containment Spray System..............................	3/4 6-13
	Spray Additive System..................................	3/4 6-14
	Containment Cooling System..............................	3/4 6-15
3/4.6.3	CONTAINMENT ISOLATION VALVES..............................	3/4 6-16
TABLE 3.6-1	CONTAINMENT ISOLATION VALVES..............................	3/4 6-18
3/4.6.4	COMBUSTIBLE GAS CONTROL	
	Hydrogen Analyzers..................................	3/4 6-30
	Hydrogen Control Systems..............................	3/4 6-31
3/4.7 PLANT SYSTEMS		
3/4.7.1	TURBINE CYCLE	
	Safety Valves..................................	3/4 7-1
TABLE 3.7-1	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING FOUR LOOP OPERATION..................	3/4 7-2
TABLE 3.7-2	(BLANK)..................................	3/4 7-2

DRAFT

INDEX

BASES

SECTION	PAGE
REACTOR COOLANT SYSTEM (Continued)	

3/4.4.5 STEAM GENERATORS.. B 3/4 4-3

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE........................ B 3/4 4-4

3/4.4.7 CHEMISTRY... B 3/4 4-5

3/4.4.8 SPECIFIC ACTIVITY...................................... B 3/4 4-5

3/4.4.9 PRESSURE/TEMPERATURE LIMITS............................ B 3/4 4-6

TABLE B 3/4.4-1 REACTOR VESSEL TOUGHNESS......................... B 3/4 4-10

FIGURE B 3/4.4-1 FAST NEUTRON FLUENCE (E>1MeV) AS A FUNCTION OF FULL POWER SERVICE LIFE........................ B 3/4 4-12

FIGURE B 3/4.4-2 EFFECT OF FLUENCE AND COPPER ON SHIFT OF RT_{NDT} FOR REACTOR VESSEL STEELS EXPOSED TO IRRADIATION AT 550°F........................ B 3/4 4-13

3/4.4.10 STRUCTURAL INTEGRITY.................................. B 3/4 4-15

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS... B 3/4 5-1

3/4.5.2, ~~and~~ 3/4.5.3, and 3/4.5.4 ECCS SUBSYSTEMS.................... B 3/4 5-1

DELETE [3/4.5.4 BORON INJECTION SYSTEM.............................. B 3/4 5-2] DELETE

3/4.5.5 REFUELING WATER STORAGE TANK........................... B 3/4 5-2

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT.................................... B 3/4 6-1

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS................... B 3/4 6-3

3/4.6.3 CONTAINMENT ISOLATION VALVES........................... B 3/4 6-4

3/4.6.4 COMBUSTIBLE GAS CONTROL................................ B 3/4 6-4

DELETE

EMERGENCY CORE COOLING SYSTEMS

3/4.5.5 BORON INJECTION SYSTEM

DRAFT

BORON INJECTION TANK

LIMITING CONDITION FOR OPERATION

3.5.5 The boron injection tank shall be OPERABLE with:

a. A minimum contained borated water volume of 900 gallons, and

b. A boron concentration of between 2000 and 2100 ppm.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With the boron injection tank inoperable, restore the tank to OPERABLE status within 1 hour or be in HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to 1% $\Delta k/k$ at 200°F within the next 6 hours; restore the tank to OPERABLE status within the next 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.5 The boron injection tank shall be demonstrated OPERABLE by:

a. Verifying the contained borated water volume at least once per 7 days, and

b. Verifying the boron concentration of the water in the tank at least once per 7 days.

DELETE

DRAFT

3/4.5.~~6~~ 5 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.~~6~~ 5 The refueling water storage tank (RWST) shall be OPERABLE with:

a. A minimum contained borated water volume of 394,000 gallons,

b. A boron concentration of between 2000 and 2100 ppm of boron,

c. A minimum solution temperature of 37°F, and

d. A maximum solution temperature of 100°F.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.~~6~~ 5 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the contained borated water volume in the tank, and

2) Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWST temperature when the outside air temperature is either less than 37°F or greater than 100°F.

DRAFT

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) accumulator ensures that a sufficient volume of borated water will be immediately forced into the core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a MODE where this capability is not required.

The requirement to verify accumulator isolation valves shut with power removed from the valve operator when the pressurizer is solid ensures the accumulators will not inject water and cause a pressure transient when the Reactor Coolant System is on solid plant pressure control.

3/4.5.2, ~~and~~ 3/4.5.3, and 3/4.5.4 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

DRAFT

ECCS SUBSYSTEMS (Continued)

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and Safety Injection pumps except the required OPERABLE charging pump to be inoperable in MODES 4 and 5 and in MODE 6 with the reactor vessel head on provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensure, that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses. The Surveillance Requirements for leakage testing of ECCS check valves ensure that a failure of one valve will not cause an intersystem LOCA. The Surveillance Requirement to vent the ECCS pump casings and accessible, i.e., can be reached without personnel hazard or high radiation dose, discharge piping ensures against inoperable pumps caused by gas binding or water hammer in ECCS piping.

3/4.5.4 BORON INJECTION SYSTEM

DELETE

The OPERABILITY of the Boron Injection System as part of the ECCS ensures that sufficient negative reactivity is injected into the core to counteract any positive increase in reactivity caused by RCS system cooldown. RCS cooldown can be caused by inadvertent depressurization, a loss-of-coolant accident, or a steam line rupture.

DELETE

The limits on injection tank minimum contained volume and boron concentration ensure that the assumptions used in the Steam Line break analysis are met. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.5.5 REFUELING WATER STORAGE TANK

The OPERABILITY of the refueling water storage tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water

Justification - Specification 3.5.5

The BIT is a component of the Safety Injection System whose sole function is to provide concentrated boric acid to the reactor coolant to mitigate the consequences of postulated steamline break accidents. Although the BIT acts to mitigate steamline breaks of various sizes occuring from any power level, the cases which serve as the Westinghouse steamline break licensing basis, and which define the existing requirements on the minimum BIT boron concentration, are as follows:

- For the "hypothetical" steamline break, i.e., double ended rupture of a main steamline, the radiation releases must remain within the requirements of 10CFR Part 100. This is the ANSI N18.2 criterion for condition IV events, "Limiting Faults." Westinghouse conservatively meets this for the SNUPPS Units by demonstrating that the DNB design basis is met, the criterion typically used for Condition II events.

- For the "credible" steamline break, i.e., the failure open of a single steam generator relief, safety, or turbine bypass valve, that radiation releases must remain within the requirements of 10CFR Part 20. This is the ANSI N18.2 criterion for Condition II events, "Faults of Moderate Frequency." Westinghouse has met this criterion by showing that the DNB design basis is met.

In order to verify that the Westinghouse criteria are met, technical specifications are applied to the BIT and associated equipment thereby assuring the validity of the safety analyses performed. Specifically, these assure that the boric acid concentration is maintained and that heat tracing is necessary to maintain the tank and associated piping at a sufficiently high temperature so that the minimum concentration requirements may be met. Furthermore, the safety-related nature of the boric acid system requires that the heating systems be redundant.

The required solubility temperature imposes a continuous load on the heaters, and low-temperature alarm actuation and heater burnout have occurred in some operating plants. Violation of the Technical Specification on concentration in the BIT poses availability problems in that recovery is required within a very short time. If the concentration is not restored within one hour, the plant must be taken to the hot standby condition and borated to the equivalent of 1 percnet $\Delta K/K$ at 200°F. Thus, this requirement has a potentially serious impact on plant availability. In addition, the boric acid concentration makes recovery from a spurious safety injection signal (which results in injection of the BIT fluid into the reactor coolant system) time consuming and costly.

These potential difficulties unfavorably affecting plant availability, operability, and maintainability can be drastically reduced in severity or eliminated by reducing the boron concentration to a minimum level at which heat tracing would no longer be required.

The only accident analyses which are affected by boron reduction, are the steamline break transients. For the SNUPPS Units, the system was analyzed assuming that the BIT remains installed, without heat tracing, and with the boric acid concentration reduced to zero ppm. This combination provides the most limiting case for tha anlayses and allows the maximum operational flexibility in implementing the possible hardward alternative.

This case is considered because it allows elimination of the presently specified heat tracing, resulting in cumulative maintenance savings on the heat tracing equipment associated with the BIT. Additionally, all Technical Specifications concerning BIT concentrations, temperatures, and associated surveillance can be eliminated, exclusive of the refueling water storage tank.

CONCLUSIONS

Plant specific analyses have been performed for the SNUPPS Units' steamline break transients and have shown that the Boron Injection Tank may be bypassed, eliminated, or reduced in boron concentration to 0 ppm and the heat tracing system deleted.

Revision 14 to the SNUPPS FSAR reflects the new analysis described above.

DRAFT

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATIONS	VALVE NUMBER	FUNCTION	TYPE LEAK TEST REQUIRED	MAXIMUM ISOLATION TIME (Seconds)
8. Hand-Operated and Check Valves - (Continued)				
P-66	EN V-017	CTMT Spray Pump B to CTMT Spray Nozzles	A	N.A.
P-45	EP V-046	Accumulator Nitrogen Supply Line	C	N.A.
P-43	HD V-016	Auxiliary Steam to Decon System	C	N.A.
P-43	HD V-017	Auxiliary Steam to Decon System	C	N.A.
P-63	KA V-039	Rx Bldg Service Air Supply	C	N.A.
P-63	KA V-118	Rx Bldg Service Air Supply	C	N.A.
P-30	KA V-204	Rx Bldg Instrument Air Supply	C	N.A.
P-67	KC V-478	Fire Protection Supply to RX Bldg	C	N.A.
P-57	SJ V-111	Liquid Sample from PASS to RCDT	A,C	N.A.
P-98	KB V-001	Breathing Air Supply to Rx Bldg	C	N.A
P-98	KB V-002	Breathing Air Supply to Rx Bldg	C	N.A.

Justification - Table 3.6-1

These changes are requested to properly account for the isolation valves associated with a recently designed and installed plant system - breathing air.

2) Verifying the fuel level in the fuel storage tank,

3) Verifying the fuel transfer pump starts and transfers fuel from the storage system to the day tank,

4) Verifying the diesel starts from ambient condition and accelerates to at least 514 rpm in less than or equal to 12 seconds.* The generator voltage and frequency shall be 4000 ± 200 volts and 60 ± 1.2 Hz within 12 seconds* after the start signal. The diesel generator shall be started for this test by using one of the following signals:

   a) Manual, or

   b) Simulated loss-of-offsite power by itself, or

   c) Safety Injection test signal.

5) Verifying the generator is synchronized, loaded to greater than or equal to 6201 kW at the maximum practical rate,* operates with a load greater than or equal to 6201 kW for at least 60 minutes, and

6) Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the day tanks;

c. At least once per 31 days by checking for and removing accumulated water from the fuel oil storage tanks;

d. At least once per 92 days and from new fuel oil prior to its addition to the storage tanks by verifying that a sample obtained in accordance with ASTM-D270-1975 meets the following minimum requirements in accordance with the tests specified in ASTM-D975-1977:

   1) A water and sediment content of less than or equal to 0.05 volume percent;

   2) A kinematic viscosity of 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes;

   3) A specific gravity as specified by the manufacturer at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89 or an API gravity at 60°F of greater than or equal to 27 degrees but less than or equal to 39 degrees;

* See insert, following page

INSERT

These diesel generator starts from ambient conditions shall be performed only once per 184 days in these surveillance tests and all other engine starts for the purpose of this surveillance testing shall be preceded by an engine prelube period and/or other warmup procedures recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.

Specification 3/4.8.1.1

Justification

This specification change was requested in order to reduce the number of "cold" starts required on the diesel generator. Diesel generator starts from ambient conditions cause a relatively large amount of stress and wear on the engine which will eventually decrease its reliability. Unnecessary wear and tear on the diesel has a negative effect on the safe operation of the reactor plant.

DRAFT

TABLE 4.8-1

DIESEL GENERATOR TEST SCHEDULE

NUMBER OF FAILURES IN LAST 100 VALID TESTS*	TEST FREQUENCY
$\leq$ 1	At least once per 31 days
2	At least once per 14 days
3	At least once per 7 days
$\geq$ 4	At least once per 3 days

*Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, Revision 1, August 1977, where the last 100 tests are determined on a per nuclear unit basis. For the purposes of this schedule, only valid tests conducted after the completion of the preoperational test requirements of Regulatory Guide 1.108 shall be included in the computation of the "last 100 valid tests."

Justification - Specification 3.8.1.1, Table 4.8-1

The requested change to the table note was added to clarify, in the Technical Specifications, the point in time at which accountability for diesel generator tests must begin.

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	TRIP SETPOINT (Amperes)	BREAKER RESPONSE TIME AT MAX. SHORT CIRCUIT (Sec/Cycles)	POWERED EQUIPMENT
480 V MOTOR CONTROL CENTER (Continued)			
P-52NG01BEF2 B-40A Fuse	75 (Inst.)	0.016 N.A.	RHR Loop Inlet Iso Vlv EJHV8701A
P-52NG03CDF4 B-15A Fuse	29 (Inst.)	0.016 N.A.	RCP Thermal Barrier CCW Iso Valve BBHV13
P-52NG03CHF1 B-15A Fuse	29 (Inst.)	0.016 N.A.	RCP Thermal Barrier CCW Iso Vlv BBHV14
P-52PG19NAF4 B-100A Fuse	400 (Inst.)	0.016 N.A.	Reactor Cavity Cooling Fan PCGN02A
P-52PG19NCF3 B-60A Fuse	260 (Inst.)	0.016 N.A.	Ctmt Atmospheric Control System Fan DCGR01A
P-52PG19NGF2 B-40 Fuse	~~None~~ 675 (Inst.)	0.017 N.A.	RCP A Space Heater
P-52PG19NGF3 B-40 Fuse	~~None~~ 675 (Inst.)	0.017 N.A.	RCP B Space Heater
P-52PG19NEF1 B-40A Fuse	170 (Inst.)	0.016 N.A.	RCP A Oil Lift Pump
P-52PG19NGR3 B-40A Fuse	170 (Inst.)	0.016 N.A.	RCP B Oil Lift Pump
P-52PG19NFF1 B-15A Fuse	22 (Inst.)	0.016 N.A.	Ctmt Normal Sump Pump DPLF05A

DRAFT

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	TRIP SETPOINT (Amperes)	BREAKER RESPONSE TIME AT MAX. SHORT CIRCUIT (Sec/Cycles)	POWERED EQUIPMENT
480 V MOTOR CONTROL CENTER (Continued)			
P-52PG19NFF2 B-15A Fuse	22 (Inst.)	0.016 N.A.	Ctmt Normal Sump pump DPLF05C
P-52PG19NAF2 B-40A Fuse	84 (Inst.)	0.016 N.A.	Instrumnt Tunnel Sump Pump DPLF07A
P-52NG03CBF4 B-15A Fuse	29 (Inst.	0.016 N.A.	RCP Thermal Barrier CCW Iso Vlv BBHV15
P-52NG03~~CB~~ CL F2 B-15A Fuse	29 (Inst.)	0.016 N.A.	RCP Thermal Barrier CCW Iso Vlv BBHV16
P-52PG20NBF5 B-100A Fuse	320 (Inst.)	0.016 N.A.	Reactor Cavity Cooling Fan DCGN02B
P-52PG20NFF4 B-60A Fuse	260 (Inst.)	0.016 N.A.	Ctmt Atmospheric Control System Fan DCGR01B
P-52PG20NBF1 B-40A Fuse	~~None~~ [illegible] (Inst.)	0.017 N.A.	RCP C Space Heater
P-52PG20NCF1 B-40A Fuse	~~None~~ [illegible] (Inst.)	0.017 N.A.	RCP D Space Heater
P-52PG20NFF3 B-40A Fuse	170 (Inst.)	0.016 N.A.	RCP C Oil Lift Pump

DRAFT

TABLE 3.8-1 (Continued)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

PROTECTIVE DEVICE NUMBER AND LOCATION	TRIP SETPOINT (Amperes)	BREAKER RESPONSE TIME AT MAX. SHORT CIRCUIT (Sec/Cycles)	POWERED EQUIPMENT
480 V MOTOR CONTROL CENTER (Continued)			
P-52PG20NFF2 B-40A Fuse	170 (Inst.)	0.016 N.A.	RCP D Oil Lift Pump
P-52PG20NER2 B-15A Fuse	22 (Inst.)	0.016 N.A.	Ctmt Normal Sump Pump DPLF05B
P-52PG20NGF4 B-15A Fuse	22 (Inst.)	0.016 N.A.	Ctmt Normal Sump Pump DPLF05D
P-52PG20NDR2 B-40A Fuse	84 (Inst.)	0.016 N.A.	Instrument Tunnel Sump DPLF07B
P-52PG1904 B-600A Fuse	1440 (Inst.)	0.03 N.A.	POLAR CRANE HKE13
CRDM CONTROL ROD DRIVE POWER			
P-10A Fuse B-30A Fuse	- - - - - -	N.A. N.A.	Gripper Coils (106 fused circuits)
P-50A Fuse B-150A Fuse	- - - - - -	N.A. N.A.	Lift Coils (53 fused circuits)

(50) - Protective Relay Instantaneous Unit
(51) - Protective Relay Inverse Time Unit
Inst. - Instantaneous Protection
S.T. - Short Time Protection

DRAFT

HEAT FLUX HOT CHANNEL FACTOR, and RCS FLOW RATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

3. The control rod insertion limits of Specification 3.1.3.6 are maintained; and

4. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions 1. through 4. above are maintained. As noted on Figure 3.2-3, RCS flow rate and $F^N_{\Delta H}$ may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if the measured $F^N_{\Delta H}$ is also low) to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

R as calculated in Specification 3.2.3 and used in Figure 3.2-3, accounts for $F^N_{\Delta H}$ less than or equal to 1.49. This value is used in the various accident analyses where $F^N_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature, and thus is the maximum "as measured" value allowed.

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction. This credit comes from generic design margins totaling 9.1% when the analysis is performed with the approved interim methods. The margin used to partially offset rod bow penalties is 9.1%. This margin breaks down as follows:

THIS PAGE OPEN PENDING RECEIPT OF INFORMATION FROM THE APPLICANT

1)	Design limit DNBR	[illegible]%
2)	Grid spacing K_s	(2.9)%
3)	Thermal Diffusion Coefficient	(1.2)%
4)	DNBR multiplier	(2.7)%
5)	Pitch Reduction	(1.7)%

The margin used to partially offset rod bow penalties is (5.9)% with the remaining (3.2)% used to trade off against measured flow which may be as much as (2)% lower than thermal design flow plus uncertainties.

The penalties applied to $F^N_{\Delta H}$ to account for rod bow as a function of burnup are consistent with those described in Mr. John F. Stolz's (NRC) letter to T. M. Anderson (Westinghouse) dated April 5, 1979 with the difference being due to the amount of margin each unit uses to partially offset rod bow penalties.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the Incore Detector Flux Mapping System, and a 3% allowance is appropriate for manufacturing tolerance.

INSERT

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the generic margin. The generic design margins, totaling 9.1% DNBR, completely offset any rod bow penalties. This margin includes the following:

1) Design limit DNBR of 1.30 vs. 1.28
2) Grid Spacing (K_s) of 0.046 vs. 0.059
3) Thermal Diffusion Coefficient of 0.038 vs. 0.059
4) DNBR Multiplier of 0.86 vs. 0.88
5) Pitch reduction

The applicable values of rod bow penalties are referenced in the FSAR.

DRAFT

BASES

3/4.4.4 RELIEF VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs minimizes the undesirable opening of the spring-loaded pressurizer Code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

[illegible] following page

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a reactor-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that reactor-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness. Results from WCAP-10043 have been used to establish plugging limit.

INSERT

Unscheduled inservice inspections are performed on each steam generator following; 1) primary to secondary tube leaks; 2) a seismic occurrence greater than the Operating Basis Earthquake; 3) a loss of coolant accident requiring actuation of the Engineered Safeguards, which for this specification is defined to be a break greater than that equivalent to the severance of a 1" inside diameter pipe, or, for a main steamline or feedline, a break greater than that equivalent to a steam generator safety valve failing open; to ensure that steam generator tubes retain sufficient integrity for continued operation. Transients less severe than these do not require inspections because the resulting stresses are well within the stress criteria established by Regulatory Guide 1.121 which unplugged steam generator tubes must be capable of withstanding.

SAFETY LIMIT VIOLATION (Continued)

c. The Safety Limit Violation Report shall be submitted to the Commission, the NSRB and the Vice President-Nuclear within 14 days of the violation; and

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A, of Regulatory Guide 1.33, Revision 2, February 1978;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Section 7.1 of Generic Letter 82-33;

c. Plant Security Plan implementation;

d. Radiological Emergency Response Plan implementation;

e. PROCESS CONTROL PROGRAM implementation,

f. OFFSITE DOSE CALCULATION MANUAL implementation, and

g. Quality Assurance Program for effluent and environmental monitoring.

6.8.2 Each procedure and administrative policy of Specification 6.8.1 above, and changes thereto, including temporary changes shall be reviewed prior to implementation as set forth in Specification 6.5 above.

6.8.3 The plant Administrative Procedures and changes thereto shall be reviewed in accordance with Specification 6.5.1.6 and approved in accordance with Specification 6.5.3.1. The associated implementing procedures and changes thereto shall be reviewed and approved in accordance with Specification 6.5.3.1.

6.8.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the recirculation portion of the Containment Spray System, Safety Injection System, Chemical and Volume Control System, and RHR System. The program shall include the following:

1) Preventive maintenance and periodic visual inspection requirements, and

Justification - Specification 6.8.1.b

The requested change was made to clarify the applicable section of Generic Letter 82-33.