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PROPOSED REVISIONS

TECHNICAL SPECIFICATIONS

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT		TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1.	Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Emergency Ventilation, Start Standby Diesel Generators, Reactor Containment Fan Coolers, and Essential Cooling Water).					
a.	Manual Initiation	2	1	2	1, 2, 3, 4	19
b.	Automatic Actuation Logic	2	1	2	1, 2, 3, 4	14
c.	Actuation Relays	3	2	3	1, 2, 3, 4	14
d.	Containment Pressure--High-1	3	2	2	1, 2, 3, 4	15
e.	Pressurizer Pressure--Low	4	2	3	1, 2, 3#	20
f.	Compensated Steam Line Pressure-Low	3/steam line	2/steam line any steam line	2/steam line in each steam line	1, 2, 3#	15
~~g.~~	~~Compensated T_{COLD} - Low-Low (interlocked with P-15)~~	~~3/loop~~	~~2/loop in any loop~~	~~2/loop in each loop~~	~~1####, 2, 3#~~	~~15~~

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation					
a. Manual Initiation					
1) Individual	2/steam line	1/steam line	2/operating steam line	1, 2, 3	24
2) System	2	1	2	1, 2, 3	23
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	22
c. Steam Line Pressure - Negative Rate--High	3/steam line	2/steam line any steam line	2/steam line in each steam line	3###	15
d. Containment Pressure - High-2	3	2	2	1, 2, 3	15
e. Compensated Steam Line Pressure - Low	3/steam line	2/steam line any steam line	2/steam line in each steam line	1, 2, 3#	15
~~f. Compensated T_{COLD} - Low-Low (interlocked with P-15)~~	~~3/loop~~	~~2/loop any loop~~	~~2/loop in each loop~~	~~1####, 2, 3#~~	~~15~~

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	25
b. Steam Generator Water Level--High-High (P-14)	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. in each operating stm. gen.	1, 2, 3	20
c. Deleted ~~Compensated T_{COLD} - Low (interlocked with P-15)~~	~~3/loop~~	~~2/loop in any loop~~	~~2/loop in each loop~~	~~1####, 2, 3#~~	~~15~~
d. Deleted ~~Feedwater Flow - High (interlocked with P-15) coincident with either of the following in 2 of 4 loops:~~	~~3/stm. gen.~~	~~2/stm. gen. in any stm. gen.~~	~~2/stm. gen. in each stm. gen.~~	~~1####, 2, 3~~	~~15~~
~~Reactor Coolant Flow-Low~~	~~3/loop~~	~~2/loop in any loop~~	~~2/loop in each loop~~	~~1, 2, 3~~	~~15~~
~~or~~					
~~T_{avg} - Low~~	~~1/loop~~	~~1/loop in any loop~~	~~1/loop~~	~~1, 2, 3~~	~~15~~
e. Safety Injection	See Item 1. for all Safety Injection initiating functions and requirements.				
f. T_{avg}-Low coincident with Reactor Trip (P-4)**	4 (1/loop)	2	3	1, 2, 3	20

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
8. Loss of Power					
a. 4.16 kV ESF Bus Undervoltage-Loss of Voltage	4/bus	2/bus	3/bus	1, 2, 3, 4	20
b. 4.16 kV ESF Bus Undervoltage-Tolerable Degraded Voltage Coincident with SI	4/bus	2/bus	3/bus	1, 2, 3, 4	20
c. 4.16 kV ESF Bus Undervoltage - Sustained Degraded Voltage	4/bus	2/bus	3/bus	1, 2, 3, 4	20
9. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	21
b. Low-Low T_{avg}, P-12	4	2	3	1, 2, 3	21
c. Reactor Trip, P-4	2	1	2	1, 2, 3	23
~~d. Power Range Neutron Flux Input to Excessive Cooldown Protection, P-15~~	~~4~~	~~2~~	~~3~~	~~1, 2, 3~~	~~21~~

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TABLE 3.3-3 (Continued)

TABLE NOTATIONS

**Feedwater Isolation only.

***Function is actuated by either actuation train A or actuation train B. Actuation train C is not used for this function.

****Automatic switchover to containment sump is accomplished for each train using the corresponding RWST level transmitter.

#Trip function may be blocked in this MODE below the P-11 (Pressurizer Pressure Interlock) Setpoint.

##During CORE ALTERATIONS or movement of irradiated fuel within containment.

###Trip function automatically blocked above P-11 and may be blocked below P-11 when ~~Excessive Cooldown Protection~~ Low Compensated Steamline Pressure Protection is not blocked.

~~####Trip function is blocked in MODE 1 above the P-15 (Excessive Cooldown Protection) setpoint.~~

ACTION STATEMENTS

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 16 - (Not Used)

ACTION 17 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is met. One additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ACTION 18 - With less than the Minimum Channels OPERABLE requirement, operation may continue provided the containment purge supply and exhaust valves are maintained closed.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWAL.
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Emergency Ventilation, Start Standby Diesel Generators, Reactor Containment Fan Coolers, and Essential Cooling Water)					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	N.A.
c. Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
d. Containment Pressure--High 1	3.6	0.71	2.0	≤ 3.0 psig	≤ 4.0 psig
e. Pressurizer Pressure--Low	13.1	10.71	2.0	≥ 1850 psig##	≥ 1842 psig#
f. Compensated Steam Line Pressure-Low	13.6	10.71	2.0	≥ 735 psig	≥ 714.7 psi
~~g. Compensated T_{COLD}-Low-Low (interlocked with P-15)~~	~~4.5~~	~~0.5~~	~~1.0~~	~~≥ 532°F~~	~~≥ 528°F***~~
2. Containment Spray					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	N.A.
c. Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
d. Containment Pressure--High-3	3.6	0.71	2.0	≤ 9.5 psig	≤ 10.5 psig

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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 VA
4. Steam Line Isolation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
c. Steam Line Pressure - Negative Rate--High	2.6	0.5	0	≤ 100 psi	≤ 126.3 psi*
d. Containment Pressure - High-2	3.6	0.71	2.0	≤ 3.0 psig	≤ 4.0 psig
e. Compensated Steam Line Pressure - Low	13.6	10.71	2.0	≥ 735 psig	≥ 714.7 psig
f. Compensated T_{COLD} - Low-Low (interlocked with P-15)	4.5	0.5	1.0	≥ 532°F	≥ 528°F***
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
b. Steam Generator Water Level--High-High (P-14)	4.5	2.35	2.0+0.2#	≤ 87.5% of narrow range instrument span.	≤ 88.9% of narrow range instrument span.
c. Deleted ~~Compensated T_{COLD}-Low (interlocked with P-15)~~	~~4.5~~	~~0.5~~	~~1.0~~	~~≥ 538°F~~	~~≥ 534°F***~~

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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 VA
5. Turbine Trip and Feedwater Isolation (Continued)					
d. Deleted ~~Feedwater Flow-High (interlocked with P-15) Coincident With Either of the Following in 2 of 4 Loops:~~	~~7.2~~	~~2.76~~	~~4.0~~	~~$\leq$ 30.0% Flow~~	~~$\leq$ 32.2% Flow~~
~~RCS Flow-Low~~	~~4.0~~	~~3.19~~	~~0.6~~	~~$\geq$91.8% of loop design flow****~~	~~$\geq$ 90.9% of loop design flow****~~
~~or~~					
~~T_{avg}-Low~~	~~4.5~~	~~1.36~~	~~0.8~~	~~$\geq$ 574°F~~	~~$\geq$ 571.1°F~~
e. Safety Injection	See Item 1 above for all Safety Injection Trip Setpoints and Allowable Values.				
f. T_{avg}-Low Coincident with Reactor Trip (P-4) (Feedwater Isolation Only)	4.5	1.36	0.8	$\geq$ 574°F	$\geq$ 571.1°F
6. Auxiliary Feedwater					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	N.A.
c. Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
d. Steam Generator Water Level--Low-Low	15.0	12.75	2.0+0.2#	$\geq$ 33.0% of narrow range instrument span.	$\geq$ 31.5% of narrow range instrument span.
e. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				

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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 VA
9. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	$\leq$ 1985 psig	$\leq$ 1993 psig
b. Low-Low T_{avg}, P-12	N.A.	N.A.	N.A.	$\geq$ 563°F	$\geq$ 560.1°F
c. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	N.A.
~~d. Power Range Neutron Flux Input to Excessive Cooldown Protection, P-15~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~< 10% Rated Thermal Power~~	~~< 12.3% Rate Thermal Powe~~
10. Control Room Ventilation					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.
b. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
c. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.
d. Control Room Intake Air Radioactivity - High	3.7×10^{-5} µCi/cc	2.2×10^{-5} µCi/cc	1.6×10^{-5} µCi/cc	$\leq 6.1\times10^{-5}$ µCi/cc	$\leq 7.8\times10^{-5}$ µCi/cc
e. Loss of Power	See Item 8. above for all Loss of Power Trip Setpoints and Allowable Values.				
11. FHB HVAC					
a. Manual Initiation	N.A.	N.A.	N.A.	N.A.	N.A.

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TABLE 3.3-4 (Continued)

TABLE NOTATIONS

*Time constants utilized in the lead-lag controller for Steam Line Pressure-Low are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

**The time constant utilized in the rate-lag controller for Steam Line Pressure-Negative Rate-High is greater than or equal to 50 seconds. CHANNEL CALIBRATION shall ensure that this time constant is adjusted to this value.

~~***Time constants utilized in the lead-lag controller for Compensated T_{COLD} are $\tau_1 \geq 12$ seconds and $\tau_2 \leq 3$ seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.~~

****Loop design flow = 95,400 gpm

#2.0% span for Steam Generator Level; 0.2% span for Reference Leg RTDs

##Until resolution of the Veritrak transmitter uncertainty issue, the trip setpoint will be set at ≥ 1869 psig, with the allowable value at ≥ 1861 psig.

###This setpoint value may be increased up to the equivalent limits of Specification 3.11.2.1 in accordance with the methodology and parameters of the ODCM during containment purge or vent for pressure control, ALARA and respirable air quality considerations for personnel entry.

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure--Low	
a. Safety Injection (ECCS)	$\leq 27^{(1)}/12^{(5)}$
1) Reactor Trip	$\leq 2^{(3)}$
2) Feedwater Isolation	$\leq 12^{(3)}$
3) Phase "A" Isolation	$\leq 33^{(1)}/23^{(2)}$
4) Containment Ventilation Isolation	N.A.
5) Auxiliary Feedwater	≤ 60
6) Essential Cooling Water	$\leq 62^{(1)}/52^{(2)}$
7) Reactor Containment Fan Coolers	$\leq 38^{(1)}/28^{(2)}$
8) Control Room Ventilation	$\leq 72^{(1)}/62^{(2)}$
9) Start Standby Diesel Generators	≤ 12
4. Deleted ~~Compensated T_{COLD}-Low-Low~~	
~~a. Safety Injection (ECCS)~~	~~N.A.~~
~~1) Reactor Trip~~	~~N.A.~~
~~2) Feedwater Isolation~~	~~N.A.~~
~~3) Phase "A" Isolation~~	~~N.A.~~
~~4) Containment Ventilation Isolation~~	~~N.A.~~
~~5) Auxiliary Feedwater~~	~~N.A.~~
~~6) Essential Cooling Water~~	~~N.A.~~
~~7) Reactor Containment Fan Coolers~~	~~N.A.~~
~~8) Control Room Ventilation~~	~~N.A.~~
~~9) Start Diesel Generators~~	~~N.A.~~
~~b. Steam Line Isolation~~	~~N.A.~~
5. Compensated Steam Line Pressure--Low	
a. Safety Injection (ECCS)	$\leq 22^{(4)}/12^{(5)}$
1) Reactor Trip	$\leq 2^{(3)}$
2) Feedwater Isolation	$\leq 12^{(3)}$
3) Phase "A" Isolation	$\leq 33^{(1)}/23^{(2)}$
4) Containment Ventilation Isolation	N.A.
5) Auxiliary Feedwater	≤ 60
6) Essential Cooling Water	$\leq 62^{(1)}/52^{(2)}$
7) Reactor Containment Fan Coolers	$\leq 38^{(1)}/28^{(2)}$

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ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
12. Loss of Power (Continued)	
c. 4.16 kV ESF Bus Undervoltage (Sustained Degraded Voltage)	$\leq$ 65
13. RCB Purge Radioactivity-High	
a. Containment Ventilation Isolation (48-inch lines)	$\leq 73^{(2)}$
b. Containment Ventilation Isolation (18-inch lines)	$\leq 23^{(2)}$
14. ~~Compensated~~ $\cancel{T_{cold}}$ ~~- Low~~	
~~a. Turbine Trip~~	~~N.A.~~
~~b. Feedwater Isolation~~	~~N.A.~~
15. ~~Feedwater Flow - High Coincident with 2 of 4 Loops Having Either Reactor Coolant Flow - Low or~~ $\cancel{T_{avg}}$ ~~- Low~~	
~~a. Turbine Trip - Reactor Trip~~	~~N.A.~~
~~b. Feedwater Isolation~~	~~N.A.~~
16. T_{avg} - Low Coincident with Reactor Trip	
Feedwater Isolation	N.A.
17. Control Room Intake Air Radioactivity - High	
Control Room Ventilation	$\leq 78^{(2)}$
18. Spent Fuel Pool Exhaust Radioactivity - High	
FHB HVAC Emergency Startup	$\leq 42^{(2)}$

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TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	DIGITAL OR 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, Feedwater Isolation, Control Room Emergency Ventilation, Start Standby Diesel Generators, Reactor Containment Fan Coolers, and Essential Cooling Water)								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic	N.A.	N.A.	N.A.	N.A.	M(1)	N.A.	N.A.	1, 2, 3, 4
c. Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.	M(6)	Q(4,5)	1, 2, 3, 4
d. Containment Pressure-High-1	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4
e. Pressurizer Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
f. Compensated Steam Line Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
~~g. Compensated T_{COLD}-Low-Low (interlocked with P-15)~~	~~S~~	~~R~~	~~M~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~1, 2, 3~~

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	DIGITAL OR 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								
e. Compensated Steam Line Pressure-Low	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
f. Compensated T_{COLD} - Low-Low (interlocked with P-15)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
5. Turbine Trip and Feedwater Isolation								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(6)	Q(4)	1, 2, 3
b. Steam Generator Water Level-High-High (P-14)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Deleted ~~Compensated T_{COLD}-Low (interlocked with P-15)~~	~~S~~	~~R~~	~~M~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~1, 2, 3~~
d. Deleted ~~Feedwater Flow-High (interlocked with P-15)~~ Coincident with either of the following in 2 of 4 loops: Reactor Coolant Flow-Low or T_{avg}-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. T_{avg}-Low Coincident with Reactor Trip (P-4) (Feedwater Isolation Only)	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	DIGITAL OR 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. 4.16 kV ESF Bus Undervoltage (Tolerable Degraded Voltage Coincident with SI)	N.A.	R	N.A.	M	N.A.	N.A.	N.A.	1, 2, 3, 4
c. 4.16 kV ESF Bus Undervoltage (Sustained Degraded Voltage)	N.A.	R	N.A.	M	N.A.	N.A.	N.A.	1, 2, 3, 4
9. 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. Low-Low T_{avg}, P-12	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Reactor Trip, P-4	N.A.	N.A	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
~~d. Power Range Neutron Flux Input to Excessive Cooldown Protection, P-15~~	~~N.A.~~	~~R(2)~~	~~M(3)~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~1, 2, 3~~
10. Control Room Ventilation								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	All

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	DIGITAL OR ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLAN[CE] IS REQUIRE[D]
11. FHB HVAC (Continued)								
c. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
d. Spent Fuel Pool Exhaust Radio-activity-High	S	R	M	N.A.	N.A.	N.A.	N.A.	With irradiated fuel in spent fuel pool.

TABLE NOTATION

(1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(2) Deleted ~~Neutron detectors may be excluded from CHANNEL CALIBRATION.~~

(3) Deleted ~~With Rated Thermal Power greater than or equal to the P-15 interlock setpoint, the ANALOG CHANNEL OPERATIONAL TEST shall consist of verifying that the P-15 interlock is in the required state by observing the permissive annunicator window.~~

(4) Except relays K807, K814, K829 (Train B only), K831, K845, K852 and K854 (Trains B and C only) which shall be tested at least once per 18 months during refueling and during each COLD SHUTDOWN exceeding 24 hours unless they have been tested within the previous 92 days.

(5) Except relay K815 which shall be tested at indicated interval only when reactor coolant pressure is above 700 psig.

(6) Each actuation train shall be tested at least every 92 days on a STAGGERED TEST BASIS. Testing of each actuation train shall include master relay testing of both logic trains. If an ESFAS instrumentation channel is inoperable due to failure of the Actuation Logic Test and/or Master Relay Test, increase the surveillance frequency such that each train is tested at least every 62 days on a STAGGERED TEST BASIS unless the failure can be determined by performance of an engineering evaluation to be a single random failure.

*During CORE ALTERATIONS or movement of irradiated fuel within containment.

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REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

The Engineered Safety Features Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates Turbine trip via P-16, closes main feedwater valves on T_{avg} below Setpoint, prevents the opening of the main feedwater valves which were closed by a Safety Injection or High Steam Generator Water Level, ~~or Excessive Cooldown Protection signal,~~ allows Safety Injection block so that components can be reset or tripped, ~~and actuates P-15.~~

Reactor not tripped - prevents manual block of Safety Injection.

P-11 On increasing pressurizer pressure, P-11 automatically reinstates Safety Injection actuation on low pressurizer pressure or ~~excessive cooldown signals,~~ Insert Ⓐ reinstates steamline isolation on ~~excessive cooldown signals,~~ and opens the accumulator discharge isolation valves. On decreasing pressure, P-11 allows the manual block of Safety Injection actuation on low pressurizer pressure or ~~excessive cooldown signals,~~ Insert Ⓐ allows the manual block of steamline isolation on ~~excessive cooldown signals,~~ and enables steam line isolation on high negative steam line pressure rate.

P-12 On increasing reactor coolant loop temperature, P-12 automatically provides an arming signal to the Steam Dump System. On decreasing reactor coolant loop temperature, P-12 automatically removes the arming signal from the Steam Dump System.

P-14 On increasing steam generator water level, P-14 automatically trips the turbine and the main feedwater pumps, and closes all feedwater isolation valves and feedwater control valves.

~~P-15 When the reactor is tripped (P-4) or when below the power range neutron flux setpoint, P-15 is present and allows Safety Injection actuation and main steamline isolation on Low-Low~~ $\cancel{T_{cold}}$ ~~and allows feedwater isolation and turbine trip from Low Compensated~~ $\cancel{T_{cold}}$ ~~or high feedwater flow.~~

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated action will be initiated when the radiation level monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance. The radiation monitors for plant operations sense radiation levels in selected plant systems and locations and determine whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents and abnormal conditions. Once the required logic combination is completed, the system sends actuation signals to initiate alarms or automatic isolation action and actuation of Emergency Exhaust or Ventilation Systems.

Ⓐ Low Compensated Steamline Pressure signals

ATTACHMENT 3

PROPOSED REVISIONS

FSAR

TABLE 1.3-2 (Continued)

SIGNIFICANT DESIGN CHANGES

Item	References FSAR	Description of Change
Containment isolation for steam generator blowdown and sample lines	Sections 7.3, 6.2.4	Containment isolation phase A signal changed to AFW initiation signal (SI or low-low SG water level).
Electrical penetration space ventilation system actuation	Section 7.3	System is no longer actuated by control room emergency ventilation signal, only by SI signal.
Actuated equipment lists	Section 7.3	Various changes as required to support system design changes.
Radiation input to containment ventilation isolation	Section 7.3	Input via redundant safety-grade RCB purge isolation monitors.
Low compensated Steamline pressure ~~Excessive cooldown protection~~ block permissive	Section 7.3	Block permissive for low compensated steamline pressure S.I. ~~excessive cooldown protection~~ changed from P-12 (low-low T_{avg}) to P-11 (pressurizer pressure). 45
Wide range RCS pressure outside containment.	Sections 7.4, 7.6	Addition of 2 RCS wide range pressure channels. Relocation of 3 transmitters outside containment.
Post accident monitoring	Section 7.5, Appendix 7B	Upgrade instrumentation to address RG 1.97, Rev. 2. instrumentation
Qualified Display Processing System (QDPS)	Section 7.5	Addition of safety-related display processing system which provides redundant data acquisition and display. System provides Class 1E control of the SG power operated relief valves, head vent throttle valves, AFW flow regulating valves, and ECW throttle valve for the essential chillers. The system also performs SG reference leg temperature compensation and RCS hot leg temperature averaging calculations. 55
Emergency Response Facilities Data Acquisition and Display	Section 7.5	Provides signal processing and display for Emergency Response Facilities and addresses SPDS requirement of NUREG0696.

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This transient is conservatively defined as an umbrella case to cover occurrence of several events of the same general nature. These include:

1. Inadvertent opening of an FW control valve.

2. Turbine overspeed (110 percent) with an open FW control valve.

3. Small steam break with an open FW control valve.

The excessive FW flow transient results from inadvertent opening of an FW
control valve when the plant is at hot shutdown and the SG is in the no-load
condition. The FW, Condensate, and Heater Drains Systems are in operation.
The stem of an FW control valve has been assumed to fail, with the valve imme-
diately reaching the full-open position. The FW flow to the affected loop is 30
assumed to step from essentially zero flow to the value determined by the
system resistance and the developed head of all operating FW pumps, with no
main feedwater flow to the other loops. Steam flow is assumed to remain at 30
zero, and the temperature of the FW entering the SG is conservatively assumed
to be 32°F. ~~Feedwater flow is isolated on a reactor coolant low Tcold signal~~;
~~a subsequent low-low Tcold signal actuates the Safety Injection System (SIS)~~.
Auxiliary Feedwater (AFW) flow, initiated by the SI signal, is assumed to
continue, with all pumps discharging into the affected SG. It is also
assumed, for conservatism in the secondary side analysis, that AFW flows to
the SGs not affected by the malfunctioned valve, in the so-called "unfailed
loops". Plant conditions stabilize at the values reached in 600 seconds, at 30
which time AFW flow is terminated. The plant is then either taken to cold
shutdown or returned to the no-load condition at a normal heatup rate with the
AFWS under manual control.

For design purposes, this transient has been assumed to occur 30 times during the 40-year life of the plant.

3.9.1.1.8 <u>Emergency Conditions</u>: The following primary system transients have been considered emergency conditions:

1. Small Loss-of-Coolant Accident (LOCA)

2. Small steam line break 41

3. Complete loss of flow

3.9.1.1.8.1 <u>Small Loss-of-Coolant Accident</u> - For design transient purposes, the small LOCA is defined as a break equivalent to the severance of a 1-in.-inside-diameter branch connection. (Breaks smaller than 0.375 in. inside diameter can be handled by the normal makeup system and produce no significant fluid systems transients.) Breaks which are much larger than 1 in. will cause accumulator injection soon after the accident and are regarded as faulted conditions. For design purposes, it is assumed that this transient occurs five times during the life of the plant. It should be assumed that the ECCS is actuated immediately after the break occurs and subsequently delivers water at a minimum temperature of 32°F to the RCS.

However, for those modes of operation when water solid operation may still be possible, procedures will further highlight precautions that minimize the severity of, or the potential for, an overpressurization transient. The following precautions of measures are considered in developing operating procedures: |54

a. Whenever the plant is water solid and the reactor coolant pressure is being maintained by the low pressure letdown control valve, letdown flow normally bypasses the normal letdown orifices.

b. If all reactor coolant pumps have stopped for more than 5 minutes during plant heatup after filling and venting has been completed and the reactor coolant temperature is greater than the charging and seal injection water temperature, a steam bubble will be formed in the pressurizer prior to restarting a reactor coolant pump. This precaution minimizes the pressure transient when the pump is started and the cold water previously injected by the charging pumps is circulated through the warmer reactor coolant components. The steam bubble will accommodate the resultant expansion is the cold water rapidly warmed.

c. If the reactor coolant pumps are stopped and the RCS is being cooled down by the residual heat exchangers, a nonuniform temperature distribution may occur in the reactor coolant loops. Prior to restarting a reactor coolant pump, a steam bubble will be formed in the pressurizer or an acceptable temperature profile will be demonstrated.

d. During plant cooldown, all steam generators will normally be connected to the steam header to assure a uniform cooldown of the reactor coolant loops. 53

These special precautions back-up the normal operational mode of maximizing periods of steam bubble operation so that cold overpressure transient prevention is continued during periods of transitional operations. These precautions do not apply to reactor coolant system hydrostatic testing.

The specific plant configurations of emergency core cooling system testing and alignment will also highlight procedural recommendations to prevent developing cold overpressurization transients. During these limited periods of plant operation, the following precautions/measures are considered in developing the procedures:

a. To preclude inadvertent emergency core cooling system actuation during heatup and cooldown, procedures required blocking the pressurizer pressure, and ~~excessive cooldown protection~~ signal actuation logic below the P-11 setpoint. low compensated steamline pressure S.I.

b. During further cooldown, closure and power lockout of the accumulator isolation valves will be performed at 1,000 psig. When the RCS temperature is reduced to or below 350°F, a maximum of one centrifugal charging pump and one HHSI pump is allowed operable by Technical Specifications. The LHSI pump does not impact the COMS analysis because of the low shutoff head (approximately 315 psi). |54

c. The recommended procedure for periodic emergency core cooling system pump performance testing will be to test the pumps during normal power operation or at hot shutdown conditions. This precludes any potential for developing a cold overpressurization transient.

6.2.1.4 Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures Inside the Containment. Following a postulated main steam line break or a main feedwater line break inside the Containment, the contents of one SG will be released to the Containment. Most of the contents of the other SGs will be isolated by the main steam isolation valves (MSIVs) and main feedwater isolation valves. Containment pressurization following a secondary side rupture depends on how much of the break fluid enters the Containment atmosphere as steam. Main steam line break flows can be pure steam or two-phase, while main feedwater line break flows are two-phase. With a pure steam blowdown, all of the break flow enters the Containment vapor space atmosphere. With two-phase blowdown, part of the liquid in the break flow boils off in the Containment and is added to the vapor space atmosphere, while the remaining liquid falls to the sump and contributes nothing to Containment pressurization. For main steam line break cases with large break area, steam cannot escape fast enough from the two-phase region of the ruptured SG, and the two-phase level rises rapidly to the steam line nozzle. A two-phase blowdown results. The duration of this blowdown is short, therefore reducing primary-to-secondary heat transfer, and the break flow is largely liquid.

For main steam line break cases with small break areas, steam can escape fast enough from the two-phase region of the SG with the ruptured line that the level swell does not reach the steam line nozzle, and a pure steam blowdown results. Because of the pressure-reducing effects of active and passive Containment heat sinks, the highest peak Containment pressure resulting from a main steam line break for a given set of initial SG conditions occurs for that case where the break area is the maximum at which a pure steam blowdown can occur. For conservatism, the main steam line break analysis assumed only pure steam blowdown for all break sizes and power levels.

Main steam line isolation is initiated on the following signals: high-2 Containment pressure, low steamline pressure ~~or low-low~~ $\cancel{T_{cold}}$ (above P-11 setpoint), high negative steamline pressure rate (below the P-11 setpoint), and manual. 57
Main feedwater line isolation is initiated by SG High-High water level, ~~excessive cooldown protection signal~~, reactor trip in conjunction with low T_{avg}, and SI. 2
Both the MSIVs and the main feedwater isolation valves are fully closed in 5 seconds.

The Auxiliary Feedwater System functions automatically following a secondary system line break to assure that a heat sink is always available to the RCS by supplying cold feedwater to the SGs. 49
For conservatism, it was assumed that the Auxiliary Feedwater System attains full flow to the SG immediately following feedwater isolation. In addition, the analysis includes the flashing of the volume of fluid located between the main Feedwater isolation valve and the affected SG. This fluid then flows through the affected SG and into the Containment. 49

The feedwater enters the SG in the two-phase region; therefore, main feedwater line break cases always result in two-phase blowdowns through smaller size lines and do not produce peak Containment pressures as severe as main steam line break cases.

To permit a determination of the effect of main steam line break upon Containment pressure, a spectrum of break sizes was assumed to occur inside the Containment, downstream from the integral steam line flow restrictors and up stream of the MSIVs. Unrestricted critical flow from the rupture was assumed. 14

7.1.2.1.8 Diversity: Functional diversity has been designed into the system. Functional diversity is discussed in Reference 7.1-1. The extent of the diverse system variables has been evaluated for a wide variety of postulated accidents. Generally, two or more diverse protection functions would automatically terminate an accident before unacceptable consequences could occur. 20

Regarding the ESFAS for a Loss-of-Coolant Accident (LOCA), a SI signal can be obtained manually or by automatic initiation from two diverse parameter measurements: 43 2

1. Low pressurizer pressure

2. High Containment pressure (HI-1)

For a steam line break accident, diversity of SI actuation is provided by: 43

~~1. Low-low compensated T_{cold}~~ 12

1. ~~2.~~ Low compensated steam line pressure

2. ~~3.~~ Low pressurizer pressure

3. ~~4.~~ For a steam line break inside Containment, high Containment pressure (HI-1) provides an additional parameter for generation of the signal. 43

All of the above sets of signals are redundant and physically separated and meet the requirements of IEEE 279-1971.

7.1.2.1.9 Bistable Trip Setpoints: Three values applicable to reactor trip and ESF actuations are specified:

1. Safety limit

2. Limiting value

3. Nominal setpoint

The safety limit is the value assumed in the accident analysis and is the least conservative value.

The limiting value is the Technical Specification value and is obtained by subtracting a safety margin from the safety limit. The safety margin accounts for instrument error, process uncertainties such as flow stratification and transport factor effects, etc.

The nominal setpoint is the value set into the equipment and is obtained by subtracting allowances for instrument drift from the limiting value. The nominal setpoint allows for the normal expected instrument setpoint drifts so that the Technical Specification limits will not be exceeded under normal operation.

TABLE 7.1-2 (Continued)

PLANT COMPARISON*

REACTOR TRIP SYSTEM (Continued)	DIFFERENCES FROM COMANCHE PEAK NUCLEAR STATION
8. Source Range Flux Detector Energization (Figure 7.2-3)	8. On Comanche Peak, each source range flux detector is energized and de-energized by logic output from a single train (the two detectors are on separate trains). On STP, to de-energize each detector, outputs from both A and B actuation trains are used; to energize each detector, output from either actuation train (A or B) is used.
ENGINEERED SAFETY FEATURES ACTUATION SYSTEMS	44
1. ~~Interlock P-15 (Figure 7.2-4)~~ Deleted	1. Deleted ~~Comanche Peak does not provide a P-15 signal. On STP, the P-15 signal is developed from P-4 (reactor tripped) or 2/4 power range detectors showing neutron flux below setpoint. P-15 is used as an interlock in excessive cooldown protection logic (Figure 7.2-9)~~.
2. Steam Generator High-High Water Level Signal (Figure 7.2-7)	2. Four channels are used for each SG (2/4 logic) on STP; three channels are used for each SG (2/3 logic) on Comanche Peak.
3. Main Steam Line Isolation Initiation (Figure 7.2-8)	3. Automatic actuation signals for Comanche Peak are high negative steam pressure rate, low steamline pressure and Containment HI-2 pressure. 57 ~~In addition to these same signals, automatic steamline isolation signals for STP include a low-low T_{cold} signal~~. 57 57

TABLE 7.1-2 (Continued)

PLANT COMPARISON*

ENGINEERED SAFETY FEATURES ACTUATION SYSTEMS (Continued)	DIFFERENCES FROM COMANCHE PEAK NUCLEAR STATION
4. Safety Injection Initiation through ~~Excessive Cooldown Protection~~ (Figures 7.2-8 and 7.2-9) low compensated steamline pressure	4. Automatic SI actuation signals used on both plants are Containment HI-1 pressure, ~~and~~ low pressurizer pressure. ~~Comanche Peak also uses~~ and low steamline pressure (2/3 in any loop). ~~STP uses the excessive cooldown signal of low-low compensated T_{cold} (2/3 in each loop, interlocked with P-15) or low steamline pressure (2/3 in each loop) in any loop~~.
5. Containment Spray Actuation (Figure 7.2-8)	5. On Comanche Peak, the spray pumps are started by the SI signal, while the containment spray signal confirms pump start and opens system valves. On STP, the SI signal does not actuate any containment spray equipment; only the containment spray signal actuates Containment Spray System equipment. 44
6. Radiation Signal Inputs to Containment Ventilation Isolation (Figures 7.3-2A and 7.2-8)	6. On Comanche Peak, the radiation inputs to the containment ventilation isolation signal are the three detectors (particulate, iodine, gas) of the containment air monitor. On STP, the radiation inputs are the two Class 1E RCB purge isolation monitors (gas detectors) and the non-Class 1E Containment atmosphere monitor (particulate, iodine, gas detectors).
7. Control Room HVAC ESF Actuation Signals (Figures 7.2-8 and 7.3-24)	7. Both plants utilize the SI signal for control room air cleanup filtration. Comanche Peak has a common control room; each control room inlet radiation monitor actuates the corresponding control room HVAC train. Also each unit's plant vent stack wide range gas radiation monitor

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TABLE 7.1-2 (Continued)

PLANT COMPARISON*

ENGINEERED SAFETY FEATURES ACTUATION SYSTEMS (Continued)	DIFFERENCES FROM COMANCHE PEAK NUCLEAR STATION	
	actuates one control room HVAC train. STP has a separate control room for each unit. Each control room has redundant air inlet radiation monitors, each actuating all three trains of control room HVAC.	
8. Fuel Handling Building Exhaust HVAC ESF Actuation Signals (Figure 7.3-27)	8. STP uses SI signal or high radiation signal (from either of two redundant Class 1E spent fuel pool exhaust monitors) to initiate FHB exhaust filtration. On Comanche Peak, fuel building exhaust is always filtered; no actuation is required.	44
9. Deleted	9. Deleted	61
10. ~~Excessive Cooldown Feedwater Isolation Signals (Figures 7.2-5, 7.2-7, 7.2-9 and 7.2-14)~~ Deleted	Deleted 10. ~~Addition for STP of excessive cooldown signals for feedwater isolation on:~~ ~~a. [low primary loop flow or low~~ ~~T_{avg}~~ ~~] in 2/4 loops + high FW flow + P-15 interlock signal, or~~ ~~b. low compensated~~ ~~T_{cold}~~ ~~+ P-15 interlock signal.~~ ~~Comanche Peak does not have these signals.~~	44
11. Turbine Trip Signal From Feedwater Isolation Signals (Figure 7.2-14)	11. Addition on STP of manual reset capability for the turbine trip signal from the combined signal of P-16 or any of the following signals: safety injection or P-14 signal. ~~or excessive cooldown feedwater isolation.~~ Comanche Peak does not provide this capability.	

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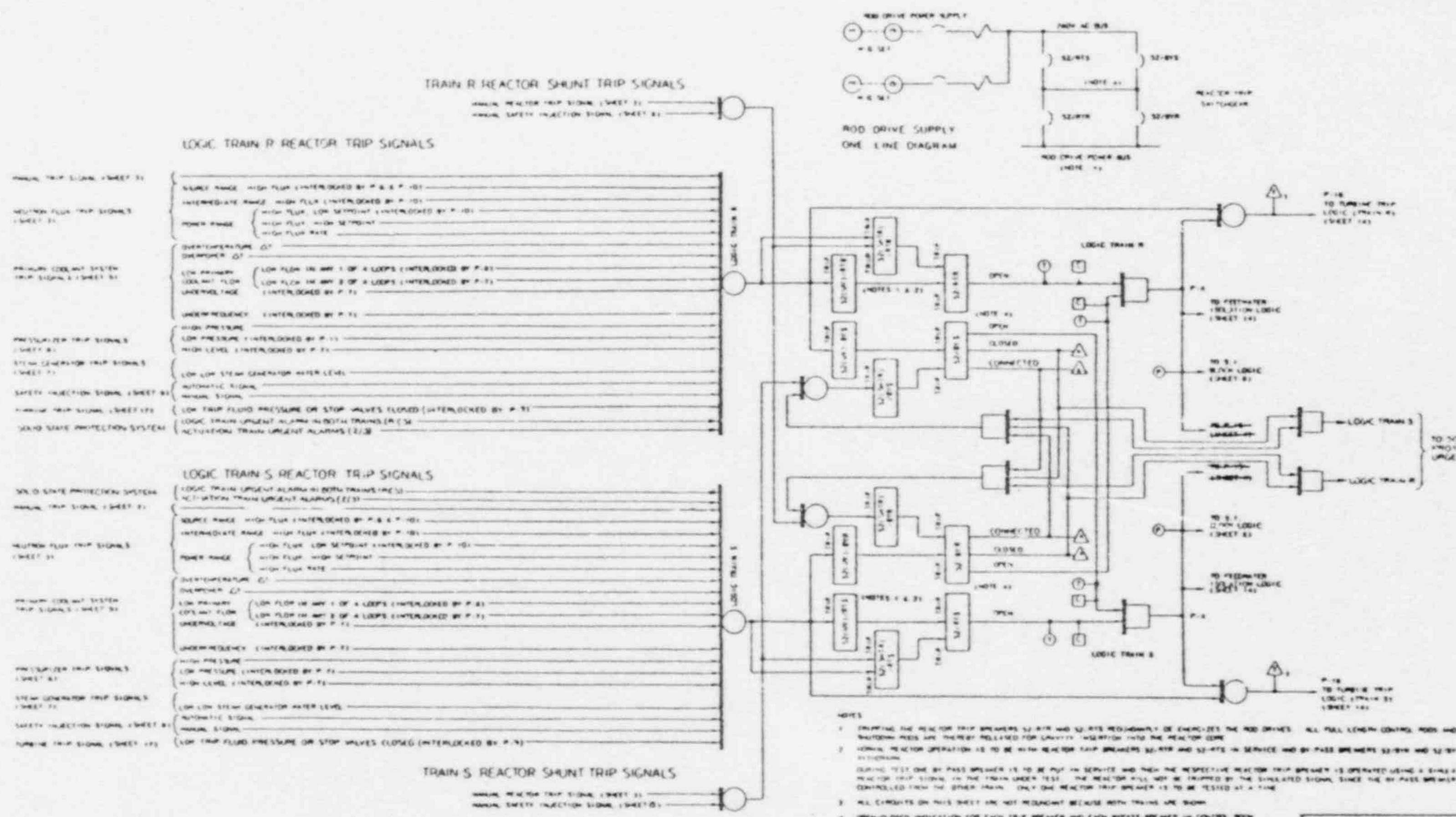
TABLE 7.1-2 (Continued)

PLANT COMPARISON*

ENGINEERED SAFETY FEATURES ACTUATION SYSTEMS (Continued)	DIFFERENCES FROM COMANCHE PEAK NUCLEAR STATION
12. P-4 Signal/Safety Injection or P-14 Signal ~~or Excessive Cooldown~~ Feedwater Isolation Interface (Figure 7.2-14)	12. Comanche Peak: After P-14 signal or SI signal is received, feedwater isolation signal is sent. This signal is then sealed in through coincidence with the P-4 reactor trip. STP: The SI signal or P-14 signal ~~or excessive cooldown~~ FW isolation signal sets a retentive memory for FW isolation. Absence of a P-4 reactor trip then allows reset of the memory.
13. P-4 Signal/Low T_{avg} Signal Feedwater Isolation Interface (Figure 7.2-14)	13. Comanche Peak: Presence of P-4 reactor trip and low T_{avg} signals sets a retentive memory (with actuation block). Manual reset of this memory allows repositioning of all FW control and bypass control valves (if closed by that signal). 44 STP: Presence of P-4 reactor trip and low T_{avg} signals seals in the low T_{avg} signal, sends a (non-resettable) closure signal to the FW control valves and sets a retentive memory (with actuation block), which can be manually reset to allow repositioning of the FW bypass control valves. (Difference is that the STP FW control valves cannot be repositioned until the reactor trip signal is removed.)
14. Auxiliary Feedwater System Actuation (Figure 7.2-16)	14. Comanche Peak: Two motor-driven pumps are automatically actuated by SI signal or blackout (LOOP) signal or trip of both main feed pumps or low-low water level in any SG. One turbine-driven pump



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SOUTH TEXAS PROJECT
UNITS 1 & 2

FUNCTIONAL DIAGRAM
REACTOR TRIP SIGNALS
(SHEET 2)

Figure 7.2-2

Amendment 5

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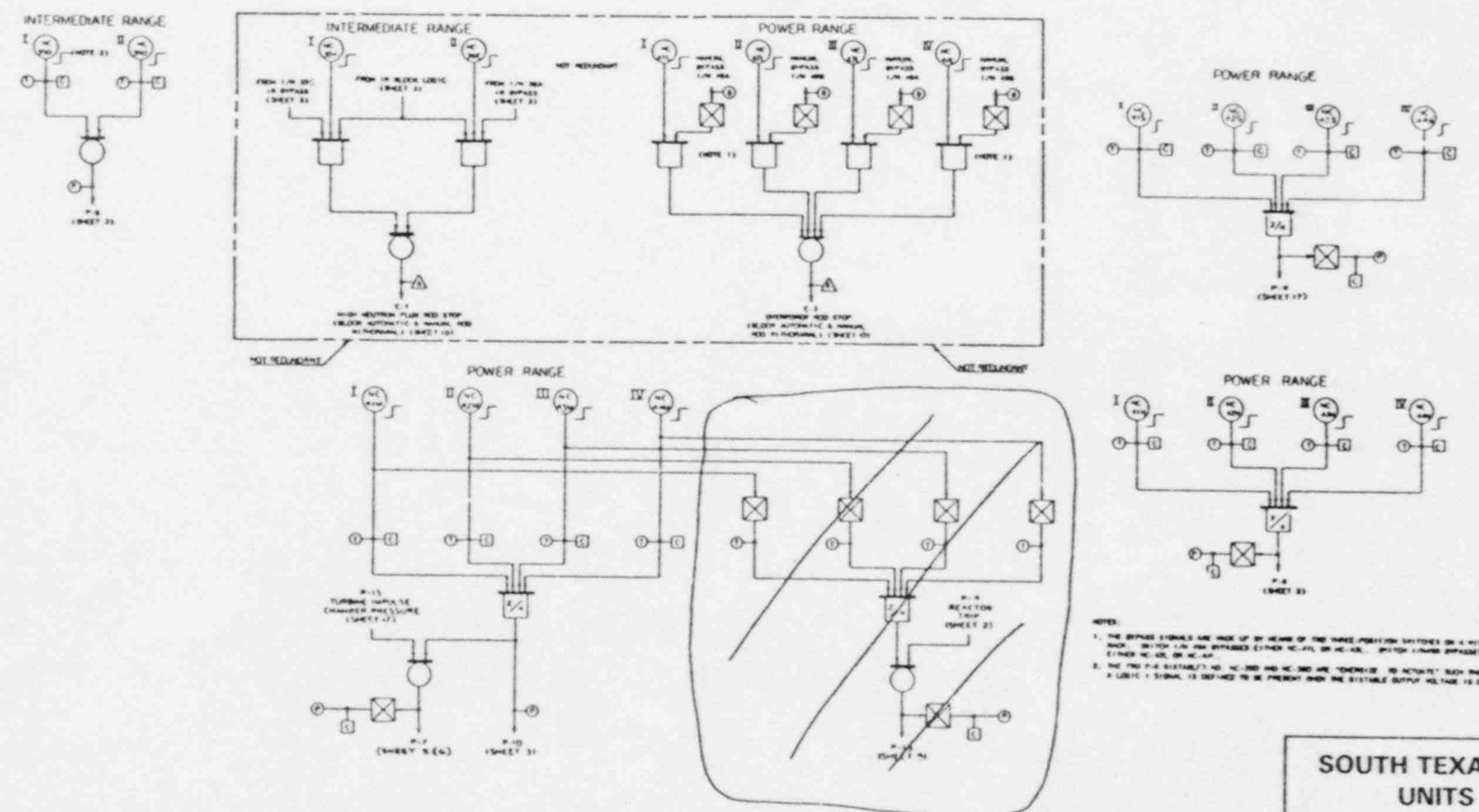

SOUTH TEXAS PROJECT
UNITS 1 & 2

FUNCTIONAL DIAGRAM
NUCLEAR INSTRUMENTATION
PERMISSIVE AND BLOCKS
(SHEET 4)

Figure 7.2-4

Amendment 57

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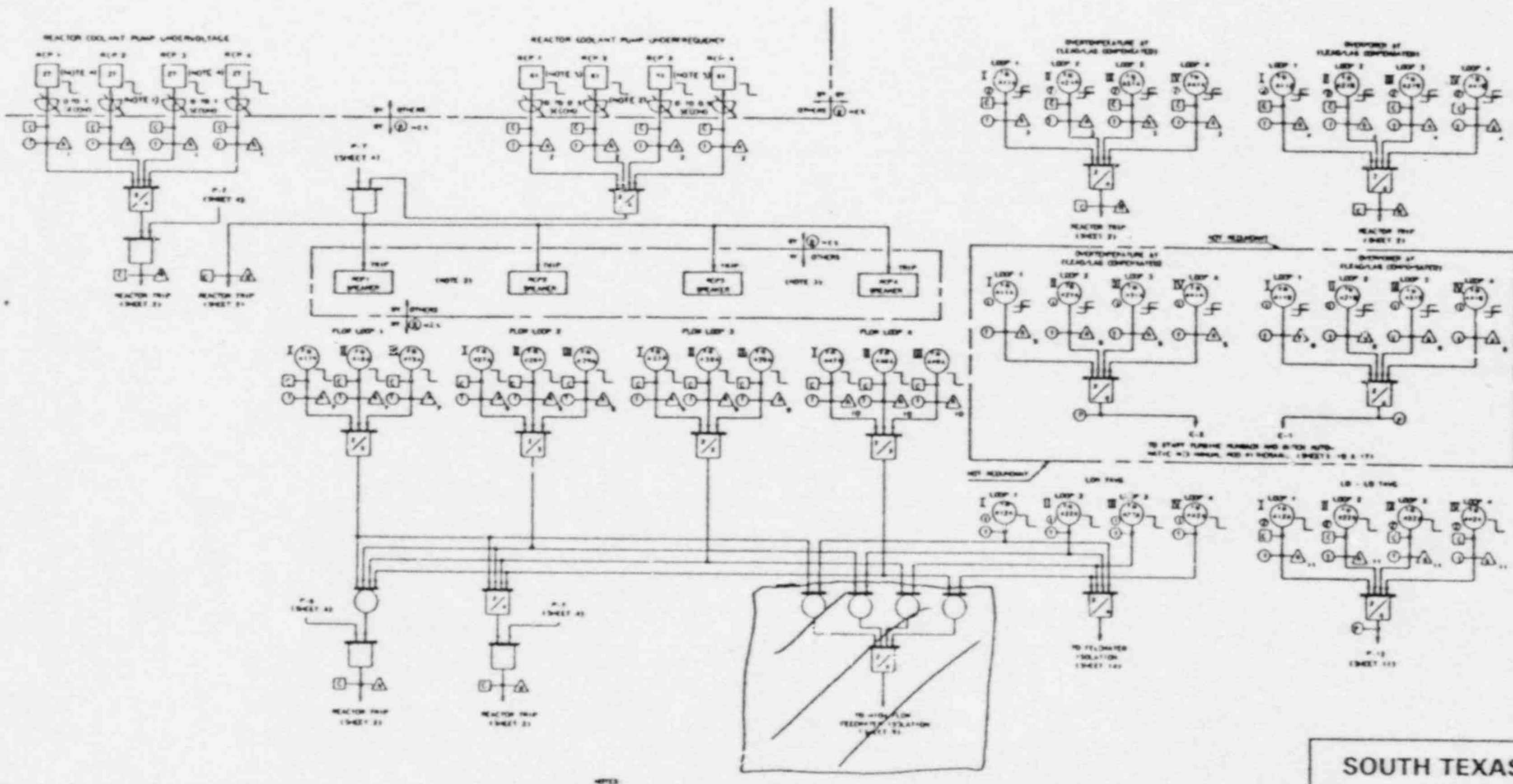


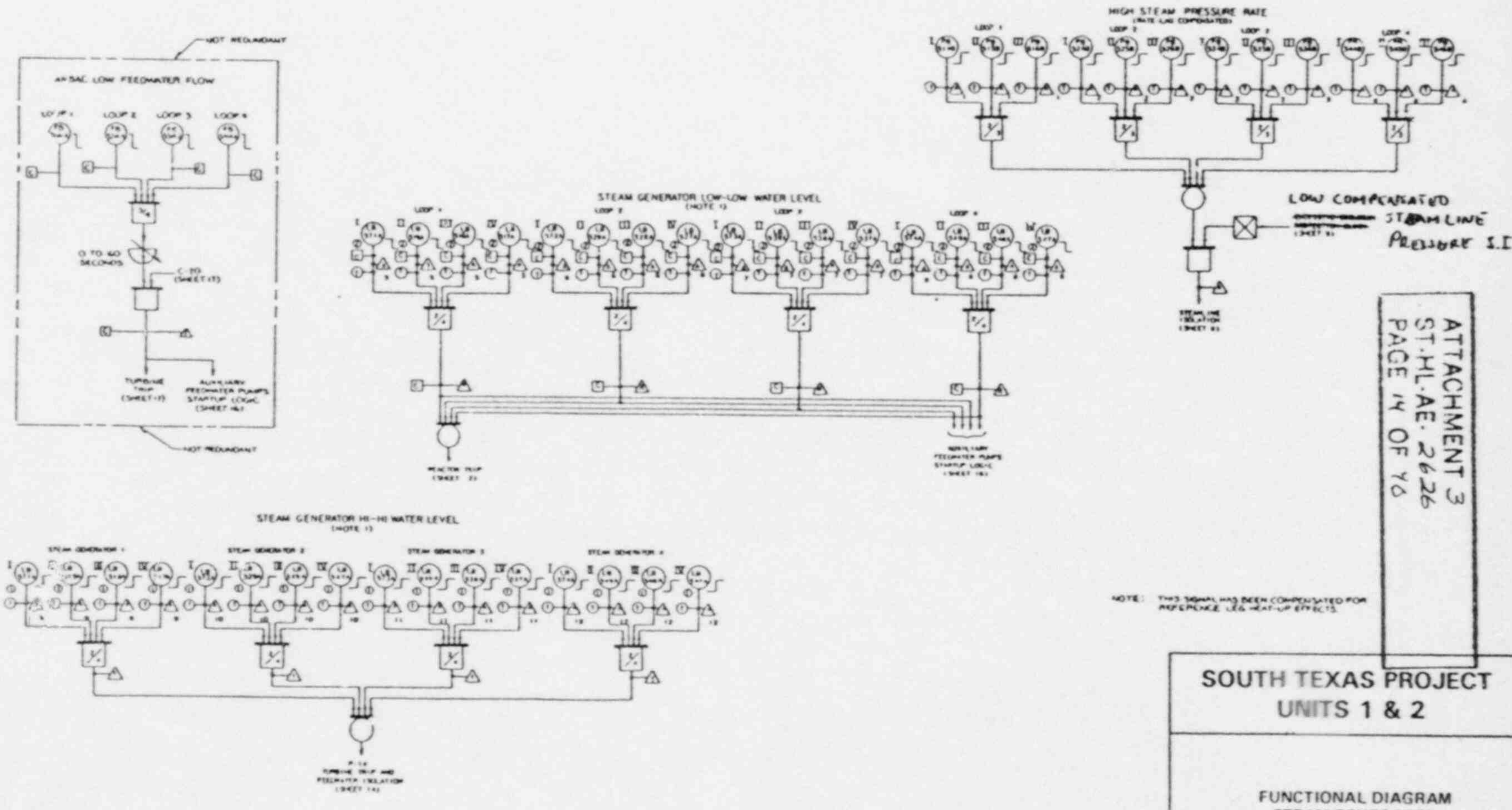
NOTES:

SOUTH TEXAS PROJECT
UNITS 1 & 2

FUNCTIONAL DIAGRAM
PRIMARY COOLANT SYSTEM
TRIP SIGNALS
(SHEET 5)

Figure 7.2-5

Amendment 57



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SOUTH TEXAS PROJECT
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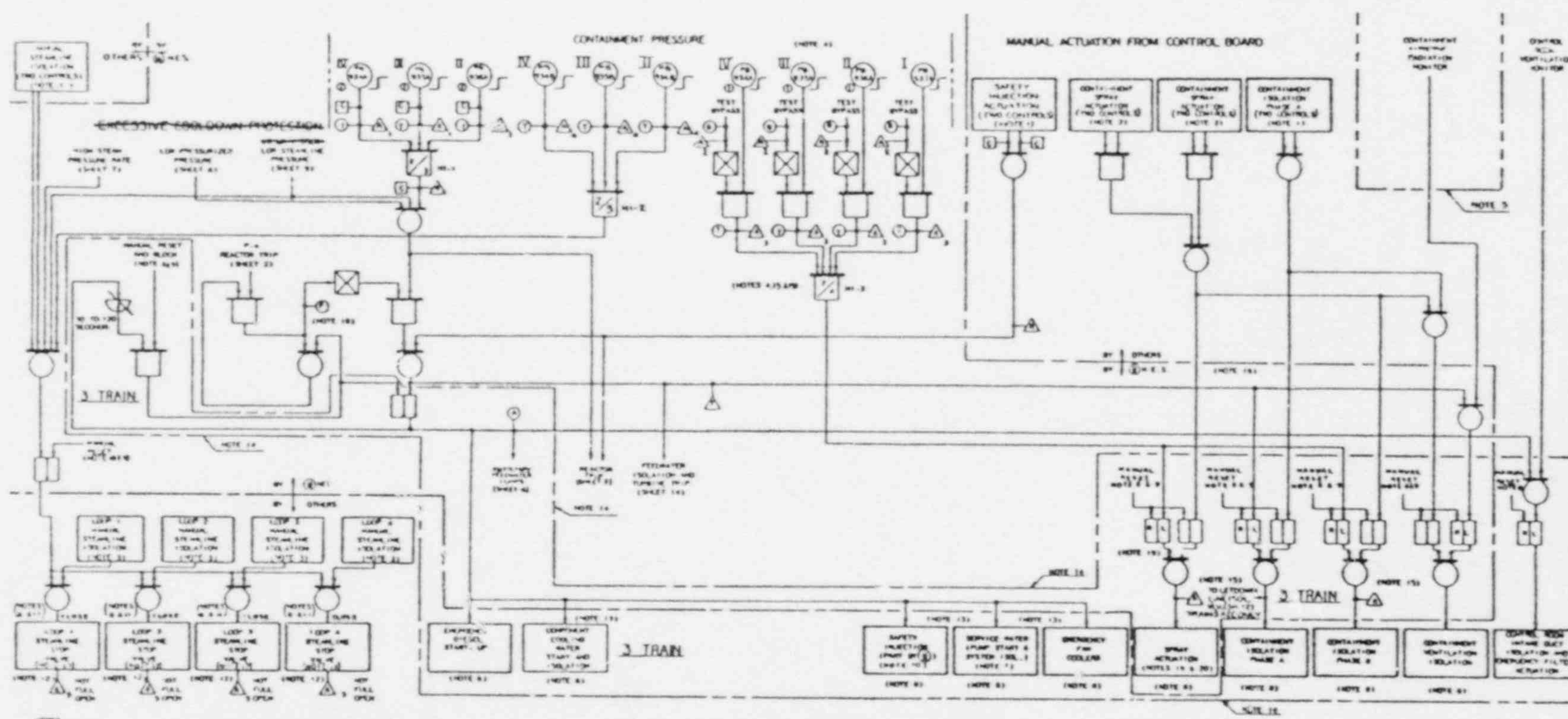
FUNCTIONAL DIAGRAM
STEAM GENERATOR
TRIP SIGNALS
(SHEET 7)

Figure 7.2-7

Amendment 57



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FUNCTIONAL DIAGRAM
SAFEGUARDS ACTUATION
SIGNALS
(SHEET 8)

Figure 7.2-8

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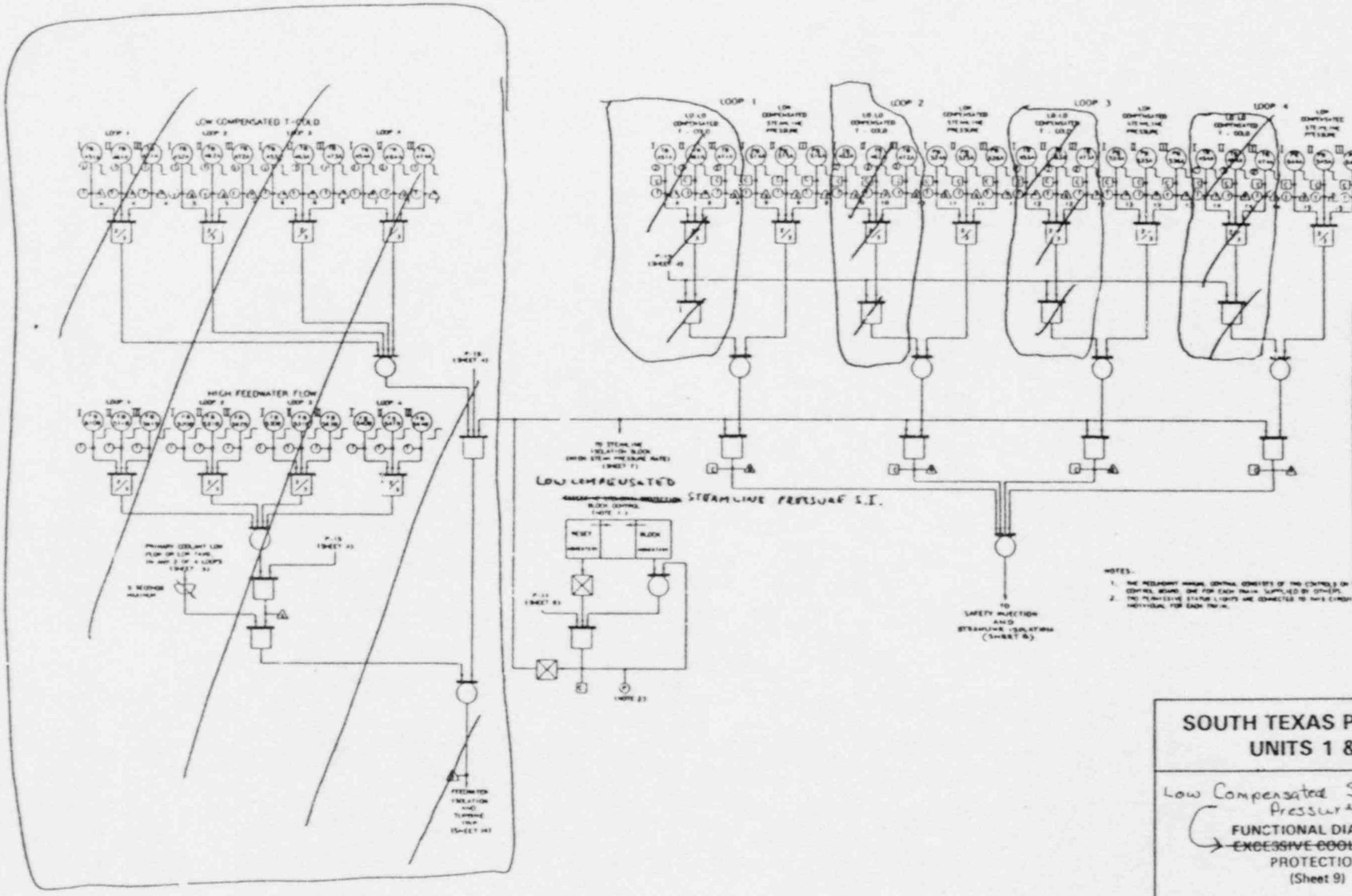

SOUTH TEXAS PROJECT
UNITS 1 & 2

Low Compensated Steamline Pressure
FUNCTIONAL DIAGRAM
~~EXCESSIVE COOLDOWN~~
PROTECTION
(Sheet 9)

Figure 7.2-9

Amendment 57

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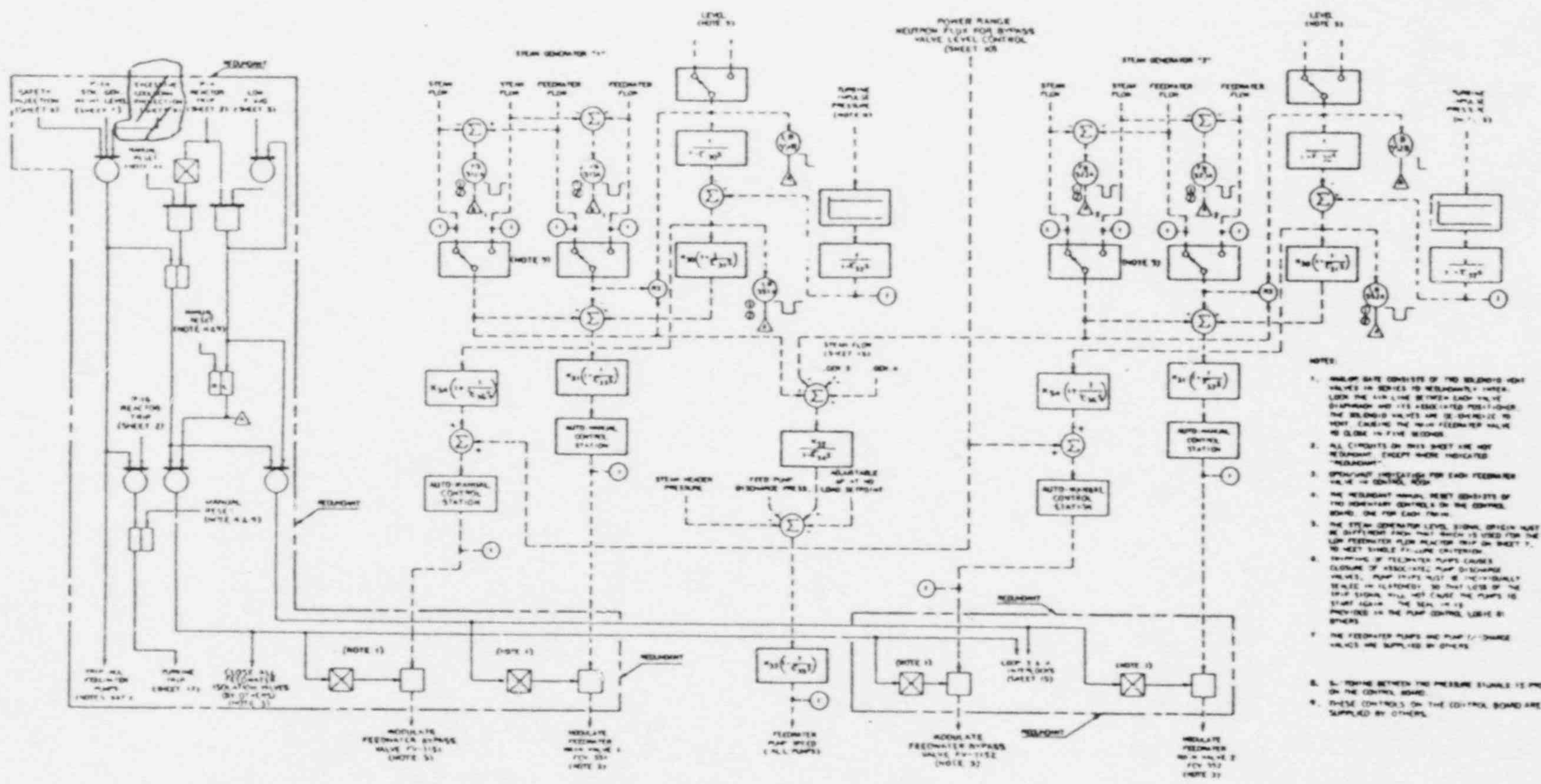

SOUTH TEXAS PROJECT
UNITS 1 & 2

FUNCTIONAL DIAGRAM
FEEDWATER CONTROL
AND ISOLATION
(SHEET 14)

Figure 7.2-14

Amendment 57

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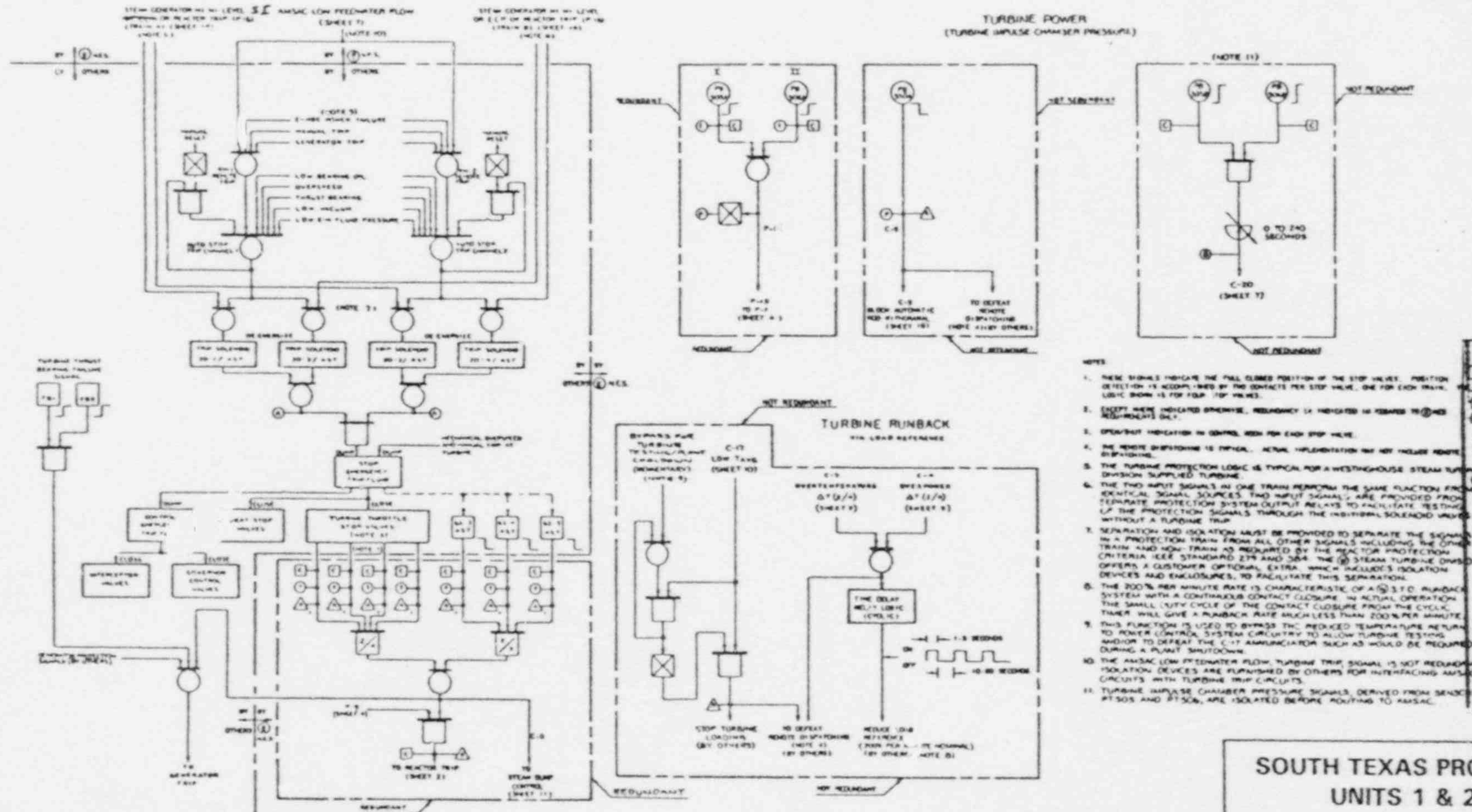

SOUTH TEXAS PROJECT
UNITS 1 & 2

TURBINE TRP, RUNBACKS,
AND OTHER SIGNALS
(SHEET 17)

Figure 7.2-17

Amendment 57

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steam pressure rate), 7.2-8 (ESF actuation), 7.2-9 (Low compensated steam line pressure protection) ~~(excessive cooldown protection)~~, 7.2-14 and 7.2-15 (feedwater control and isolation), and 7.2-16 (auxiliary feedwater).

To facilitate ESF actuation testing, six cabinets (two per train) are provided which enable operation, to the maximum practical extent, of safety features loads on a group-by-group basis until actuation of all devices has been checked. Final actuation testing is discussed in detail in Section 7.3.1.2.

7.3.1.1.4 Final Actuation Circuitry: The Solid-State Protection System supplies the following signals: 43

1. Safety injection signal (Table 7.3-5 lists actuated equipment. Typical control logics for actuated equipment are shown on Figures 7.3-2 through 7.3-8.)

43

2. Containment spray signal (Table 7.3-6 lists actuated equipment. Typical control logics for actuated equipment are shown on Figures 7.3-9 and 7.6-14.)

3. Containment isolation Phase A signal (Table 7.3-7 lists actuated equipment. Typical control logics are shown on Figures 7.3-11 through 7.3-13.) 43

4. Containment isolation Phase B signal (Table 7.3-8 lists actuated equipment. Typical control logics are shown on Figures 7.3-14 and 7.3-15.) 61

5. Containment ventilation isolation signal (Table 7.3-9 lists actuated equipment. Typical control logics are shown on Figures 7.3-16 and 7.3-17.)

6. Steam line isolation signal (Table 7.3-10 lists actuated equipment. Typical control logics are shown on Figures 7.3-18 and 7.3-18A.)

7. Feedwater isolation signal (Table 7.3-11 lists actuated equipment. Typical control logics are shown on Figures 7.3-19 and 7.3-20.) 43

8. Auxiliary Feedwater (AFW) initiation signal (Table 7.3-15 lists actuated equipment. Typical control logics are shown on Figures 7.3-21, 7.3-21A and 7.3-21B.) Q032.16 45

Loads are sequenced onto the three Class 1E ESF buses by the ESF load sequencers, as described in Chapter 8. The design meets the requirements of GDC 35. 43

7.3.1.1.5 Design Bases Information. The functional diagrams presented on Figures 7.2-5 through 7.2-9 and 7.2-14 through 7.2-16 provide a graphic outline of the functional logic associated with requirements for the ESFAS. Requirements for the ESFAS are given in Chapter 15. Given below is the design bases information required in Institute of Electrical and Electronic Engineers (IEEE) 279-1971.

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7.3.1.1.5.1 <u>Generating Station Conditions</u> - The following is a summary of those generating station conditions requiring protective action: 43

1. Primary system:

   a. Rupture in small pipes or cracks in large pipes

   b. Rupture of a reactor coolant pipe or LOCA

   c. Rupture of a SG tube 43

2. Secondary system:

   a. Minor secondary system pipe breaks resulting in steam release rates equivalent to a single dump, relief, or safety valve

   b. Rupture of a major secondary system pipe

3. Fuel handling accident inside Containment

7.3.1.1.5.2 <u>Generating Station Variables</u>: The accidents identified above are described in Chapter 15, including the ESFAS signals used to mitigate the accident consequences. The variables listed below are monitored for the automatic initiation of ESF systems during these accidents. Post-accident monitoring requirements are discussed in Section 7.5.

1. Containment pressure

2. Pressurizer pressure

3. Steam line pressure

~~4. Reactor coolant cold leg temperature (T_{cold})~~ 43

4. ~~5~~. SG water level

~~6. Feedwater flow~~

~~7. Primary coolant flow~~

5. ~~8~~. Normal and Supplementary Containment purge exhaust radiation 61

6. ~~9~~. Containment atmosphere radiation (no credit taken for this parameter)

7.3.1.1.5.3 (Paragraph Deleted) ~~Spatially Dependent Variables: The only variable monitored for deriving ESFAS accident mitigating signals that could be considered spatially dependent is the T_{cold} measurement, which is made at the cold leg of each loop downstream of the reactor coolant pump. In this location turbulent mixing at the pump will eliminate stratification~~. 43

7.3.1.1.5.4 <u>Limits, Margins, and Setpoints</u>: Prudent operational limits, available margins, and setpoints before onset of unsafe conditions requiring protective action are discussed in Chapter 15 and the Technical Specifications. 43

2. Typical maximum allowable time delays in generating the actuation signal for secondary system break protection, in addition to the above, are:

a. Steam line pressure (from which steam line pressure rate is also derived and to which add 0.5 sec)	0.6 seconds	
~~b. T_{cold} (direct immersion in cold leg)~~	~~5.0 seconds with flow 78% of nominal and straight line to 10 seconds at zero flow~~	
b ~~c~~. Actuation signals for auxiliary feedwater pumps (steam generator water level)	2.0 seconds	43
~~d. Primary loop flow~~	~~1.0 seconds~~	
~~e. Feedwater flow~~	~~2.0 seconds~~	

3. The time delay in generating the Containment ventilation signal for a fuel handling accident inside Containment is the total of the time delay in the radiation monitors and the time delay in the Solid-State Protection System to generate the Containment ventilation isolation signal. The maximum allowable time delay is 12.5 seconds for the design basis release analyzed in Section 15.7. 43 53 58

7.3.1.1.5.6.2 <u>System Accuracies</u> -

1. Typical accuracies required for generating the required actuation signals for Reactor Coolant System break protection are: 43

a. Pressurizer pressure (uncompensated)	$\pm$14 psi	
b. Containment pressure	$\pm$1.8 percent of full scale	57

2. Typical accuracies required in generating the required actuation signals for secondary system break protection, in addition to the above, are:

a. Steam line pressure	$\pm$2.5% of span	43
~~b. T_{cold}~~	~~$\pm$2°F~~	
b. ~~c~~. Actuation signals for auxiliary feedwater pumps (steam generator water level)	$\pm$2.3 percent of span	43
~~d. Primary loop flow~~	~~$\pm$2.75% ΔP span~~	13

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~~e. Feedwater flow $\pm 5.0\%$ ΔP span~~ 13

3. Typical accuracy in generating the required radiation actuation signals for the Containment ventilation isolation signal is ± 33 percent. 53

7.3.1.1.5.6.3 Ranges of Sensed Variables to be Accommodated Until Conclusions of Protective Action are Assured - 43

1. Typical ranges required in generating the actuation signals for Reactor Coolant System break protection are:

a.	Pressurizer pressure	1,700 to 2,500 psig
b.	Containment pressure	-5 to 65 psig	53

2. Typical ranges required in generating the actuation signals for secondary system break protection, in addition to the above, are:

a.	Steam line pressure (from which steam line pressure rate is derived)	0 to 1,400 psig
~~b.~~	~~T_{cold}~~	~~510° to 630°F~~	43
b. ~~c.~~	Actuation signals for auxiliary feedwater pumps (steam generator water level)	$\pm$ 6 ft from nominal full-load water level
~~d.~~	~~Primary loop flow~~	~~0 to 120% ΔP~~
~~e.~~	~~Feedwater flow~~	~~0 to 100% ΔP*~~

3. The typical range required in generating the radiation actuation signals for the Containment ventilation isolation signal is 1×10^{-6} $\mu Ci/cm^3$ to 0.1 $\mu Ci/cm^3$ 43

7.3.1.1.6 Final System Drawings. Functional block diagrams, electrical elementaries, and other drawings required to perform a safety review are listed in Section 1.7.

7.3.1.2 Analysis

7.3.1.2.1 Failure Modes and Effects Analyses. Failure modes and effects analyses have been performed generically on the ESFAS within the scope of Westinghouse and documented in Reference 7.3-4. The results verify that these systems meet protection single-failure criteria as required by IEEE 279-1971. The STP ESFAS, although not identical, is designed to equivalent 43 49

~~*Corresponds to 0 to 120 percent of rated FW flow at design rating.~~

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The output of each of the initiation circuits consists of a master relay which drives slave relays for contact multiplication as required. The master and slave relays are mounted in the ESFAS cabinets, designated Train A, Train B, and Train C respectively, for the redundant counterparts. The master and slave relay circuits operate various pump and fan circuit breakers or starters, motor-operated valve contactors, solenoid-operated valves, standby diesel generator starting equipment and other ESF actuation devices. 43 53 43

7.3.1.2.2.5.4.2 Analog Testing - Analog testing is identical to that used for reactor trip circuitry as described in Section 7.2.2.2.3 and includes the following analog channels for other safety-related circuits: 43

1. Containment pressure

2. Pressurizer pressure 43

~~3. Reactor coolant cold leg narrow range temperature (excessive cooldown protection)~~ 43

~~4. Feedwater flow (excessive cooldown protection)~~

~~5. Primary coolant flow (excessive cooldown protection)~~ 43

3. ~~6~~. Steam line pressure

An exception to this is Containment spray, which is energized to actuate 2/4 and reverts to 2/3 when one channel is in test.

7.3.1.2.2.5.4.3 Solid-State Logic Testing - Except for Containment spray channels, solid-state logic testing is the same as that discussed in Section 7.2.2.2.3. During logic testing of one train, the other logic train can initiate the required ESF function (Ref. 7.3-2). 43 53

7.3.1.2.2.5.4.4 Actuation Testing - At this point, testing of the initiation circuits through operation of the master relay and its contacts to the coils of the slave relays has been accomplished. Slave relays do not operate because of reduced voltage. 43

The ESFAS final actuation device or actuated equipment testing is performed from the Safeguards Test Cabinets. These cabinets are located adjacent to the ESFAS cabinets. There is one set of test cabinets provided for each of the three actuation trains, A, B, and C. Each set of cabinets contains individual test switches necessary to actuate the slave relays. To prevent accidental actuation, test switches are of the type that must be rotated and then depressed to operate the slave relays. Assignments of contacts of the slave relays for actuation of various final devices or actuators have been made so that groups of devices or actuated equipment can be operated individually during plant operation without causing plant upset or equipment damage. In the unlikely event that an SI signal is initiated during the test of the final device that is actuated by this test, the device will already be in its safeguards position. 43 2

During this last procedure, close communication between the main control room operator and the operator at the test panel is required. Prior to the

handle unexpected events which can be better dealt with by operator appraisal
of changing conditions following an accident.

It is most important to note that manual control of the spray system does not
occur once actuation has begun by just resetting the associated logic devices
alone. Components seal in (latch) so that removal of the actuation signal, in
itself, neither cancels nor prevents completion of protective action nor 43
provides the operator with manual override of the automatic system by this
single action. In order to take complete control of the system to interrupt
its automatic performance, the operator must deliberately unlatch relays which
have "sealed in" the initial actuation signals in the associated motor control 43
center, in addition to tripping the pump motor circuit breakers, if stopping
the pumps is desirable or necessary.

The manual reset feature associated with Containment spray, therefore, does
not perform a bypass function. It is merely the first of several manual operations
required to take control from the automatic system or interrupt its
completion should such an action be considered necessary.

In the event that the operator anticipates system actuation and erroneously
concludes that it is undesirable or unnecessary and imposes a standing reset
condition in one train (by operating and holding the corresponding reset
switch at the time the actuation signal is transmitted), the other trains 43
automatically carry the protective action to completion. In the event that
the reset condition is imposed simultaneously in all three trains at the time 43
the actuation signals are generated, the automatic sequential completion of
system action is interrupted and control has been taken by the operator.
Manual takeover is maintained, even though the reset switches are released, if 43
the original actuation signal exists. Should the actuation signal then clear
and return again, automatic system actuation will repeat.

Any time delays imposed on the system action are applied after the initiating
signals are latched. In this way, delays of actuation signals for fluid 43
system lineup, load sequencing, etc., do not provide the operator additional
time to interrupt automatic completion with manual reset alone, as would be
the case if a time delay were imposed prior to sealing of the initial actuation
signal.

low compensated steamline
The manual block controls of pressurizer pressure input and ~~excessive cooldown~~ 43
pressure ~~protection~~ input to the SI signal provide the operator with the means to block
initiation of SI during plant shutdown and startup and allow main steam line
low compensated steamline pressure
isolation on high steam pressure negative rate (~~excessive cooldown protection~~ 53
block only). These block features meet the requirements of Paragraph 4.12 of
IEEE 279-1971 in that automatic removal of the block occurs when plant
conditions require the protection system to be functional.

7.3.1.2.2.7 Manual Initiation of Protective Actions (RG 1.62): There
are eight individual main steam isolation momentary control switches (two per 43
loop) mounted on the control board. Each switch, when actuated, isolates one
of the main steam lines. In addition, there are two system-level switches.
Operating either switch isolates all four steam lines at the system level.

No exception to the requirements of IEEE 279-1971 has been taken in the manual
initiation circuit of safety injection. Although Paragraph 4.17 of IEEE

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TABLE 7.3-2

INSTRUMENTATION OPERATING CONDITION FOR WESTINGHOUSE ESFAS 43

No.	Functional Unit	No. of Channels	No. of Channels To Trip	
1.	Safety Injection Signal			43
	(See Figures 7.2-8 and 7.2-9)			
	a. Manual	2	1	
	b. HI-1 Containment pressure	3	2	43
	c. Low compensated steam line pressure*	12 (3/steam line)	2/3 in any steam line	43
	d. Pressurizer low pressure*	4	2	
	~~e. Low-low compensated T_{cold}* (interlocked with P-15)~~	~~12 (3/loop)~~	~~2/3 in any loop~~	43 57
2.	Containment Spray Signal (See Figure 7.2-8)			43
	a. Manual**	2	1	
	b. Containment pressure HI-3	4	2	
3.	Auxiliary Feedwater Initiation Signal			
	(See Figure 7.2-16)			43 Q32.16
	a. Safety Injection Signal	See item 1 of this table		
	b. Steam generator low-low water level	16 (4/SG)	2/4 in any SG	

*Permissible bypass if reactor coolant pressure is less than P-11 (nominally 1900 psig). 43

**Manual actuation of Containment spray is accomplished by actuating either of two sets (two switches per set). Both switches in a set must be actuated to obtain a manually initiated spray signal. The sets are wired to meet separation and single-failure requirements of IEEE 279-1971. Simultaneous operation of two switches is desirable to prevent inadvertent spray actuation.

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TABLE 7.3-3

INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

No.	Functional Unit	No. of Channels	No. of Channels To Trip
1.	Containment Isolation Phase A (See Figure 7.2-8)		
	a. Safety Injection	See item 1 (a through e) of Table 7.3-2	
	b. Manual	2	1
2.	Steamline Isolation (See Figure 7.2-8)		
	a. High steam pressure negative rate (enabled by ~~Excessive~~ low compensated steamline pressure ~~Cooldown Protection~~ SI Block - see Figure 7.2-9)	12 (3/steamline)	2/3 in any steamline 43
	b. Low compensated steamline pressure**	12 (3/steamline)	2/3 in any steamline
	~~c. Low-low compensated T_{cold}** (interlocked with P-15)~~	~~12 (3/loop)~~	~~2/3 in any loop~~
	c. ~~d~~. Manual*	2	1
	d. ~~e~~. Containment Pressure HI-2	3	2
3.	Feedwater Line Isolation (See Figures 7.2-8 and 7.2-14)		
	a. SG hi-hi water level	16 (4/SG)	2/4 in any SG
	b. Safety Injection	See item 1 (a through e) of Table 7.3-2	
	~~c. Low compensated T_{cold} (interlocked with P-15)~~	~~12 (3/loop)~~	~~2/3 in any loop~~

* In addition to the two system level steamline isolation switches, each steam loop is provided with switches to effect steamline isolation in that loop.

~~** Permissible bypass if reactor coolant pressure is less than P-11 (nominally 1900 psig~~).

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TABLE 7.3-3 (Continued)

INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

No.	Functional Unit	Channels	No. of Channels To Trip	
~~d.~~	~~Low primary loop flow or low T_{avg} in 2/4 loops, high FW flow and P-15~~	~~See Figures 7.2-5 and 7.2-9~~		
c. ~~e.~~	Low T_{avg} (interlocked with P-4)	4 (1 per loop)	2	53
4.	Containment Isolation Phase B			
a.	Containment Spray	See item 2 (a and b) of Table 7.3-2		
5.	Containment Ventilation Isolation			
a.	Safety Injection	See item 1 (a through e) of Table 7.3-2		43
b.	Manual Containment Spray Actuation	See item 2a of Table 7.3-2		
c.	Manual Containment Isolation Phase A	See item 1b of this table		
d.	High radiation signal*	2	1	

* High radiation signal is derived from 1 of 3 radiation monitors: two Class 1E RCB Purge Isolation monitors and one Containment atmosphere monitor (non-Class 1E). High radiation signal is redundantly provided to logic trains R and S. These radiation monitors are discussed in Section 11.5.

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TABLE 7.3-4

INTERLOCKS FOR ENGINEERED SAFETY FEATURES ACTUATION SYSTEM

Designation	Input	Function Performed
P-4	Reactor tripped	Presence of P-4 signal activates turbine trip*
		Presence of P-4 signal closes main FW valves on T_{avg} below setpoint
		Presence of P-4 signal prevents opening of main FW valves which are closed by SI or high SG water level. ~~or excessive cooldown protection~~
		Presence of P-4 signal allows manual reset/block of automatic safety injection signal 43
		Absence of P-4 signal defeats the manual reset/block for safety injection
		~~Input to P-15~~
P-11	2/3 pressurizer pressure below setpoint	Presence of P-11 allows manual block of SI on low pressurizer pressure 43
		Presence of P-11 allows manual block of ~~excessive cooldown protection functions~~ low compensated steamline pressure SI (see Figure 7.2-9)
		Absence of P-11 opens all accumulator discharge isolation valves.

*P-4 is an input to P-16. The P-16 signal trips the turbine. The P-16 signal is present when either the P-4 signal is present (indicating the reactor trip circuit breaker(s) are open) or the reactor trip train-oriented logic signal is present.

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TABLE 7.3-4 (Continued)

INTERLOCKS FOR ENGINEERED SAFETY FEATURES ACTUATION SYSTEM

Description	Input	Function Performed	
P-12	2/4 T_{avg} below low-low setpoint	Presence of P-12 blocks steam dump except for cooldown condenser dump valves	43
		Presence of P-12 allows manual bypass of steam dump block for the cooldown valves only	43
P-14	2/4 SG water level above setpoint on any SG	Presence of P-14 closes all FW control and bypass valves	
		Presence of P-14 trips all main FW pumps and closes all FW isolation and bypass valves	43
		Presence of P-14 actuates turbine trip	
~~P-15~~	~~Reactor trip (P-4) or 2/4 neutron flux (power-range) below setpoint~~	~~Presence of P-15 allows SI actuation and main steam line isolation on low-low T_{cold} and allows FW isolation and turbine trip from low compensated T_{cold} or high FW flow~~	57

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10.3.2.4 <u>Power-Operated Relief Valves (PORVs)</u>. The PORVs, one for each
MS line, are required for removal of heat from the Nuclear Steam Supply System
(NSSS) during periods when the condenser is not available as a heat sink or 39
when the MSIVs are closed. The valves are ASME Class 2 and are supplied with
Class 1E power.

The design mass flowrate of each PORV (one per SG, four total) is 68,000 lb/hr 54 56
saturated steam at 100 psia. The wide open condition does not exceed 1.05 x
10^6 lb/hr at 1,300 psia. The valve design is in accordance with ASME B&PV
Code, Section III, Subsection NC and has a design pressure and temperature of 56
1,285 psig and 600°F, respectively. The operation of these valves is not
required to protect against SG overpressure or to provide the necessary safety
relief capacity.

The PORVs, which are equipped with electric-hydraulic actuators and controlled
through the Qualified Display Processing System (QDPS) discussed in Section 53
7.5.6, are set to open below the lowest SG safety valve setting to preclude
the operation of safety valves during transients when the condenser is
unavailable as a heat sink. The opening of the valves is automatic, based 39
upon steam line pressure. A remote pressure control station is provided for
each PORV to permit setpoint adjustments of each valve over the entire
pressure range up to the safety valve setting. Remote manual operation is
provided for a safe shutdown at the control room and at the auxiliary shutdown
panel. Local control is provided in case of complete loss of automatic
control. Direct position indication is provided, with input also to the QDPS 39
computer.

10.3.2.5 <u>Main Steam Isolation Valves</u>. The MSIVs are located in each MS
line downstream of the PORV, and as close to the RCB as practical (see Figure
10.3-1). A small bypass valve at each isolation valve is provided for startup
purposes.

Steam is conducted from each SG in a separate line through the RCB, each line
being anchored at the Containment wall. Main steam line anchorage is covered
in Section 3.8.1 and Containment isolation in Section 6.2.4. The lines have
the capability to absorb thermal expansion. Testing of the MSIVs is discussed 39
in Section 14.2.3.1.

The MSIVs and bypass isolation valves are provided with remote manual con- 39
trols. Automatic signals which close the MSIVs and the small bypass isolation
valves are the HI-2 Containment pressure, low steamline pressure~~, low-low~~
~~T_{cold}~~ and, the high negative steamline pressure rate signals. The MSIVs use 57
piston actuators and the bypass valves use diaphragm actuators. The valves
are held open by instrument air pressure on the bottom of the actuator.
Spring pressure on the actuator acts as the driving force for valve closure.

The MSIV logic is shown in Figure 7.3-18. To assure safety function actua-
tion, redundant actuation solenoid vent valves, powered from separate Class 1E 39
power sources, open to vent air from the bottom of the piston actuator through Q32.11
two separate vent lines. Remote valve position indications are provided in Q32.34
the control room. An annunciator located in the control room alarms on MSIV
closure.

TABLE 15.0-6

PLANT SYSTEMS AND EQUIPMENT AVAILABLE FOR
TRANSIENT AND ACCIDENT CONDITIONS

Incident	Reactor Trip Functions	ESF Actuation Functions	Other Equipment	ESF Equipment	
15.1 Increase in Heat Removal by the Secondary Systems					60
Feedwater system malfunctions causing an increase in feedwater flow	Power range high flux, overtemperature ΔT, overpower ΔT, manual	High-high steam generator water level-produced feedwater isolation and turbine trip	Feedwater isolation valves, steam generator safety valves, steam generator PORVs	Auxiliary Feedwater System	60 60
Excessive increase in secondary steam flow	Power range high flux, overtemperature ΔT, overpower ΔT, manual	-	Pressurizer safety valves, pressurizer PORVs, main steam isolation valves.		60
Inadvertent opening of a steam generator relief or safety valve	Low pressurizer pressure, safety injection signal, overtemperature ΔT, overpower ΔT, power range high flux, manual	~~Low-low compensated~~ ~~T_{cold},~~ Low pressurizer pressure, low compensated steam line pressure, manual	Feedwater isolation valves, main steam isolation valves	Auxiliary Feedwater System, Safety Injection System	43 18
Steam system piping failure	Low pressurizer pressure, safety injection signal, power range high flux, overpower ΔT, manual	~~Low-low compensated~~ ~~T_{cold},~~ Low pressurizer pressure, low compensated steam line pressure, HI-1 and HI-2 Containment pressure, manual	Feedwater isolation valves, main steam isolation valves	Auxiliary Feedwater System, Safety Injection System	3

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b. Two out of three ~~Excessive cooldown protection (2/3~~ low compensated steamline pressure ~~from any SC or 2/3 low-low compensated T-cold~~ signals in any loop) 55 2

2. Reactor trip will occur from either a) high neutron flux, b) overpower ΔT, c) two out of four low pressurizer pressure signals, or d) receipt of an SI signal. 43 57

3. Redundant isolation of the main feedwater lines: sustained high feedwater flow would cause additional cooldown. Therefore, (A) ~~an excessive cooldown protection signal, or~~ a Safety Injection signal will rapidly close all feedwater control valves and feedwater isolation valves and trip the main feedwater pumps. 57 43

4. Closure of the fast-acting main steam isolation valves (MSIVs) (designed to close in less than 5 seconds) from either a:

   a. Low compensated steamline pressure ~~or low-low T-cold~~ signal (two out of three in any loop) above the P-11 setpoint, or 57

   b. High negative steamline pressure rate signal (two out of three in any loop) below the P-11 setpoint.

A block diagram summarizing various protection sequences for safety actions required to mitigate the consequences of this event is provided in Figure 15.0-9. 2 Q211.6

Systems and equipment which are available to mitigate the effects of the accident are also discussed in Section 15.0.8 and listed in Table 15.0-6.

15.1.4.2 Analysis of Effects and Consequences.

Method of Analysis

The following analyses of a secondary system steam release are performed for this section:

1. A full plant digital computer simulation using the LOFTRAN (Ref. 15.1-1) code to determine RCS temperature and pressure during cooldown, and the effect of safety injection;

2. Analyses to determine that there is no consequential damage to the core or reactor coolant system. 2 3

The following conditions are assumed to exist at the time of a secondary system steam release: 2

1. End-of-life shutdown margin at no-load, equilibrium xenon conditions, and with the most reactive RCCA stuck in its fully withdrawn position. Operation of RCCA banks during core burnup is restricted in such a way that addition of positive reactivity in a secondary system steam release accident will not lead to a more adverse condition than the case analyzed.

(A) in addition to the normal control action which will close the main feedwater valves following a reactor trip, temperature coefficient, the cooldown results in an insertion of positive reactivity. If the most reactive RCCA is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down by the boric acid delivered by the Safety Injection System. 3

The analysis of a main steam line rupture is performed to demonstrate that the following criterion is satisfied:

Assuming a stuck RCCA, with or without offsite power, and assuming a single failure in the SIS, the core remains in place and intact. 3

Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable, the analysis, in fact, shows that no DNB occurs for any rupture assuming the most reactive assembly stuck in its fully withdrawn position.

A major steam line rupture is classified as an ANS Condition IV event (see Section 15.0.1).

The major rupture of a steam line is the most limiting cooldown transient and, thus, is analyzed at zero power with no decay heat. Decay heat would retard the cooldown thereby reducing the return to power. A detailed analysis of this transient with the most limiting break size, a double-ended rupture, is presented here.

The following functions provide the necessary protection for a steam line rupture:

1. Safety injection actuation from either: 2

   a. Two out of four low pressurizer pressure signals, or 57

   b. ~~Excessive cooldown protection (two~~ Two out of three low compensated steamline pressure ~~from any SG or two out of three low low compensated T-cold~~ in any loop~~)~~.

   c. Two out of three High-1 containment pressure signals.

2. The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the SI signal. 44

3. Redundant isolation of the main feedwater lines: sustained high feedwater flow would cause additional cooldown. Therefore, in addition to Ⓐ ~~the~~ safety injection signal, ~~an excessive cooldown protection signal~~ will rapidly close all feedwater control valves and feedwater isolation valves, as well as trip the main feedwater pumps. 43

4. Closure of the fast-acting Main Steam Isolation Valves (MSIVs) (designed to close in less than 5 seconds) from either a

   a. High-2 containment pressure signal, (two out of three) 57

   b. Low compensated steamline pressure ~~or low-low T-cold~~ signal (two out of three in any loop) above the P-11 setpoint, or

Ⓐ the normal control action which will close the main feedwater valves following a reactor trip, a

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Question 211.32

Certain automatic safety injection signals are blocked to preclude unwanted actuation of these systems during normal shutdown and startup operations. Describe the alarms available to alert the operator to a failure in the primary or secondary system during this phase of operation and the time frame available to mitigate the consequences of such an accident. Justify the time frame available.

Response

During the shutdown the following operator actions pertain to the isolation of Emergency Core Cooling System (ECCS) equipment and would effect a Loss-of-Coolant Accident (LOCA). (Start-up is not addressed since shutdown is more limiting due to the high core decay heat generation).

(i) Below the P-11 setpoint, the operator is instructed to manually block the automatic safety injection (SI) actuation circuit. This action disarms the SI signals from the pressurizer pressure transmitters and the low compensated steamline pressure transmitters ~~excessive cooldown protection logic~~. The containment high pressure signal remains armed and will actuate SI if the setpoint is exceeded. Manual SI actuation is also available. The circuit will automatically unblock if the Reactor Coolant System (RCS) pressure should increase above the P-11 setpoint.

(ii) At 1000 psig or below, the operator closes and locks out the SI accumulator discharge isolation valves.

(iii) At approximately 350 psig and 350°F, the operator aligns the Residual Heat Removal System (RHRS) for cooldown. 55

The significance of these actions on the mitigation of a LOCA are:

(i) Below the P-11 setpoint SI will be initiated by the HI-1 containment pressure signal. For small LOCAs (<2 in. diameter break) manual SI initiation may be required. The results for this event are analyzed in the safety significance portion of this question.

(ii) Between 1000 psig and 350 psig, a portion of the ECCS may be actuated automatically on containment high pressure signal or manually by the operator. The equipment that can be energized are the Low-Head and High-Head SI pumps. Three trains of SI are required to be operational in Modes 1, 2, and 3. In Mode 4 one train of SI plus one additional LHSI pump are required to be operational. The other HHSI pumps are locked out per Technical Specification requirements. However, at least one of the locked out HHSI pumps can be restored to operable status within 30 minutes. The operator would reinstitute power in the main control room to the accumulator isolation valves.

(iii) Below 350 psig, the system is in the RHRS cooling mode. The operator would manually initiate SI and isolate the RHR system from the RCS.

Response (Continued)

3. The HHSI pumps and the acumulators are locked out when the break occurs. However, operator action can be taken to unlock one of the HHSI pumps. (This is a conservative asumption for South Texas because three trains of HHSI and LHSI pumps are required operable in Mode 3.)

4. One LHSI pump is available (a second pump is assumed to fail) from either manual SI actuation or automatic actuation by the containment HI-1 signal.

For breaks of 6 and 8 inches the calculations show that one low head SI pump turned on manually by the operator 10 minutes following the break gives sufficient flow to prevent the top of the core from being uncovered. For the 8 inch break SI flow was initiated at 10 minutes plus 25 seconds (delay time between operator manually actuating safety injection and the beginning of flow). For the 6 inch break, although the SI signal was generated by the operator at 10 minutes, SI flow did not start until approximately 18 minutes following the break when the RCS pressure dropped below the LHSI pump shutoff head of 700 ft.

The RCS pressure transient for a 4 inch break is so slow that the operator, in addition to manually activating the LHSI pump at 10 minutes, is conservatively assumed to unlock one of the HHSI pumps at 30 minutes following the break. With one LHSI pump and one HHSI pump available at these times, the core remains covered.

Another facet which must be considered is the availability of alarms which would alert the operator to manually initiate SI for very small LOCAs (1-2 inch diameter) that do not pressurize the containment to containment HI-1 set pressure (5.5 psig) (which would automatically initiate safety injection). 55

The Class 1E indication available to the operator includes the narrow range water level sensors. In addition, the alarms available would include the Reactor Coolant Pressure Boundary (RCPB) leak detection system alarms. Break flow from a 1 inch break is on the order of 500 gpm and a 2 inch break would have a flow of approximately 2000 gpm. Thus, these breaks would be expected to set off the RCPB leak detection alarms much sooner than an hour after the break occurs. Based on the Inadequate Core Cooling Study (WCAP-9753) for full operation, a 1 inch break would exhibit an extremely long transient prior to core uncovery from the initiation of break flow (approximately 2.5 hrs for a 4 loop plant). Other small break analyses with SI for similar 4-loop plants were reviewed and similar results were found. An even longer transient would be expected for a small break during shutdown. Thus, the operator would have ample time to diagnose the situation, initiate SI and prevent core uncovery. For a 2 in. break, the RCPB leak detection alarms would sound within 30 minutes of initiation of the break. From McGuire low power test analyses (5 percent power), for a 2 in. break no core uncovery occurs prior to 1.67 hours. Thus, the operator again has ample time to initiate safety injection manually.

When RCS pressure is below the P-11 setpoint and SI is blocked on low pressurizer pressure and ~~excessive cooldown protection~~, a steamline rupture
low compensated steamline pressure

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Response (Continued)

would be less severe from a core integrity standpoint than the steamline ruptures at hot zero power presented in Chapter 15. Technical Specification require shutdown margins such that the return-to-power transient would be less severe than the cases presented in Chapter 15. 55

The engineered safeguards functions desired during a steamline rupture are actuation of SI and steamline isolation. When the low pressurizer pressure signals and the A ~~excessive cooldown protection~~ signals are blocked, SI and steamline isolation may be automatically initiated by the following signals: 55

1. HI-1 Negative Steamline Pressure Rate Signal

This signal is unblocked automatically when the A ~~excessive cooldown protection~~ signals are blocked. 29

(Actuates steamline isolation.)

2. Containment Pressure Signal

(Actuates SI (HI-1) & steamline isolation (HI-2).) 57

SI and steamline isolation may also be actuated manually by the operator. During a steamline break, steamline pressure, pressurizer pressure, pressurizer level and steam generator water level will tend to decrease and steam flow will increase. These parameters are all displayed in the control room. The operator's attention may be drawn to them by the following alarms; 55

a. Low pressurizer level deviation alarm

b. Low pressurizer level alarm

c. Steam flow/feedwater flow mismatch alarm

d. Low steam generator level deviation alarm

e. Low steam generator level alarm

A low compensated steamline pressure S.I.

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Response (Cont'd)

System (RCS) for xenon decay. Replenishment of the AFST may be from one of two nonsafety systems (Demineralized Water or Secondary Make-up Storage Tank) or from the Essential Cooling Pond (ECP) while boric acid is added to the RCS via the safety grade Chemical and Volume Control System (CVCS). 55

For the following events involving breaks in the RCS or secondary system piping, additional requirements for operator action have been identified.

Main Steam Line Break (See Table Q211.52-1) 41

Following the hypothetical main steam line break (MSLB) incident, a main steam isolation signal will be generated, causing the main steam isolation valves to close within ten seconds of the break. If the break is downstream of the isolation valves, all of which subsequently close, the break will be isolated. If the break is upstream of the isolation valves, or if one valve fails to close, three SGs will be isolated while one will continue to blow down. Only in the case in which one SG continues to blow down is operator action required.

In the analysis of a steam line break (Section 15.1.5), it has been assumed that the faulted steam generator is unisolable. The low steam line pressure signal automatically closes the MSIVs and initiates a SI signal which results in MFIV closure in all loops, as well as SI flow. 57 The analysis proceeds for 600 seconds (10 minutes). All applicable safety criteria are met for this event without assuming operator action. 55 It is implicitly assumed, however, that within a reasonable time period (30 minutes), the operator will begin corrective measures to orderly shutdown the plant, in accordance with the plant's Emergency Operating Procedures as discussed below.

The only source of water to the SG will be AFW since the SI signal or reactor trip signal interlocked with low Tave ~~or diverse excessive cooldown protection signal~~ will cause main feedwater isolation to occur. 55 Following main steamline isolation, steam pressure in the steam line for the unisolated SG will continue to fall rapidly, while pressure stabilizes in the remaining three main steam lines. The difference in steam line pressures, available seconds after steam line isolation, will provide the information necessary to identify the affected SG at which time the operator will isolate the AFW to the SG. Manual controls for the AFW pumps and the AFW regulating and isolation valves are provided in the control room. The required equipment for detecting the affected SG and isolating its AFW is safety grade. The operator's failure to isolate the AFW to the SG or the isolation of the wrong generator will result in the SG continuing to blowdown. 41

The second required operator action is to manually control repressurization of the RCS. Following the automatic SI actuation and after the affected SG has been isolated, continued operation of the Safety Injection System (SIS) will increase the RCS pressure to the maximum SI pump shutoff head (~1600 psia). (The RCS can be repressurized without isolating the affected SG; however the process will take longer). Above 1600 psia the operator must restore normal pressure and level control systems. The operator may terminate SI based on criteria established in the Emergency Operating Procedures. If the operator 55 fails to stop the SI pumps after the pressurizer level and pressure return to

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Question 440.01N

In response to our previous question (211.85) regarding deletion of the emergency boration system (EBS) from the STP design, you have indicated that EBS deletion was justifiable, since, in the event of a main steam line break, the DNB design bases are met and the radiation releases are within the limits set forth in 10 CFR Part 100. We have reviewed the system aspects of the revised steam line break analysis in FSAR Section 15.1.5. Based on our review we have determined that the following additional information is required. If this information has been included elsewhere in your FSAR, appropriate references in Section 15.1.5 will suffice. Likewise, if the information has been provided in the form of other documentation (e.g., Westinghouse topical reports), reference to such documentation (please be specific) is appropriate.

a. Clarification of the methodology for calculating reactivity feedback, including the effect of nonuniform core inlet temperatures from the reactor coolant loops; justification of the conservatism in the methodology with regard to the peak power obtained.

b. Clarification of the methodology used in calculating DNBR and verification that the power distributions used for DNBR calculations reflect the effect of nonuniform core inlet temperatures from the reactor coolant loops.

c. With respect to ESF actuation functions for an SLB, describe and justify the differences between the protection functions at the STP and the actuation functions in NUREG-0452, "Standard Technical Specifications for Westinghouse PWR's". Describe the "excessive cooldown protection" function, which, in accordance with the FSAR, provides safety injection in the event of an SLB. Identify the actuation set points.

Response

a) See revised Section 15.1.5.

b) See revised Section 15.1.5.

c) With respect to Engineered Safety Features (ESF) actuation functions for a SLB, the differences between the protection functions at STP and NUREG-0452 are as follows:

   1. For safety injection (SI) actuation, the function~~s~~e of ~~(a)~~ low compensated steamline pressure ~~and (b) low-low compensated T-cold coincident with P-15 have~~ has been added. The functions of ~~(c)~~ high differential pressure between steam lines and ~~(d)~~ high steam flow in two steam lines coincident with low-low T_{avg} or low steam line pressure have been deleted. 44

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2. For steamline isolation, the functions of (a) high steam flow coincident with low steamline pressure of low-low T_{avg}, and (b) high-high (HI-2) containment pressure have been deleted. In addition, the functions of (c) high steam pressure rate and (d) SI signal have been added.

This SI signal change was described to the NRC in December 1974 in Section 7.3 of RESAR-41. During CP review, the description of the ESF actuation functions referenced the RESAR-41 design and was subsequently approved by the NRC via the issuance of the STP SER, NUREG-75/075, August 1975.

~~The steamline isolation signals have been changed since issuance of the STP SER. RESAR 41 showed steamline isolation on the following conditions:~~

~~a. Containment pressure HI-2~~
~~b. Containment pressure HI-1~~
~~c. Low-low T_{avg} coincident with P-15~~
~~d. Low compensated steam line pressure~~ 44

~~As indicated above, items b, c, and d are used to generate a SI signal, with low pressurizer pressure the only other condition generating an SI signal. Since the setpoints for HI-1 and HI-2 Containment pressure were the same, it was decided to delete the HI-2 signal and replace it by the SI signal. The only actual difference is the addition of low pressurizer pressure. In addition, to provide the automatic isolation of the steam lines when SI is blocked (below P-11), the high steam pressure rate signal is used.~~

~~The "excessive cooldown protection" is comprised of the functions described in 1 (a) and (b) above. A more detailed description follows. The excessive cooldown protection logic is shown in Figure 7.2-9. Actuation of SI can be caused by either (1) two of three low-low compensated T-cold signals in any Reactor Coolant System loop coincident with P-15 or (2) two of three low compensated steamline pressure signals in any one main steamline. P-15 is caused by two of four instruments indicating a neutron flux below 10 percent power or a reactor trip.~~ 55

The actuation setpoints are given established in the Technical Specifications. 53

The excessive cooldown protection logic has been deleted from the South Texas Project protection system. This system, first described in RESAR-41, is not taken credit for in the South Texas Project plant specific licencing basis analysis.

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Response

The Main Steamline Isolation Valve (MSIV) closure logic has been modified to be consistent with that of other Westinghouse plants. The MSIV closure on manual and high steam pressure rate signals will be maintained. The MSIV closure on a safety injection signal has been modified to MSIV closure only on a HI-2 containment pressure signal and, ~~from the excessive cooldown protection logic,~~ on a low steamline pressure signal. ~~and on a low-low T_{cold} signal~~. 57 57