ATTACHMENT B

PROPOSED CHANGES TO THE TECHNICAL

SPECIFICATIONS FOR OPERATING LICENSES NPF-11 AND NPF-18

NPF-11	NPF-18
B 2-9	B 2-9
3/4 3-1	3/4 3-1
Inserts A and B	Inserts A and B
3/4 3-5	3/4 3-5
3/4 3-7	3/4 3-7
3/4 3-8	3/4 3-8
Insert C	Insert C
3/4 3-9	3/4 3-9
Insert D	Insert D
3/4 3-14	3/4 3-14
3/4 3-20	3/4 3-20
3/4 3-21	3/4 3-21
3/4 3-22	3/4 3-22
3/4 3-23	
3/4 3-25	
3/4 3-26	3/4 3-26
Insert E	Insert E
3/4 3-27	3/4 3-27
3/4 3-27(a)	3/4 3-27(a)
3/4 3-32	3/4 3-32
3/4 3-33	3/4 3-33
3/4 3-34	3/4 3-34
3/4 3-35	3/4 3-35
3/4 3-36	3/4 3-36
3/4 3-38	3/4 3-38
3/4 3-39	3/4 3-39
3/4 3-41	3/4 3-41
3/4 3-46	3/4 3-46
3/4 3-47	3/4 3-47
3/4 3-49	3/4 3-49
3/4 3-50	3/4 3-50
Insert F	Insert F
3/4 3-52	3/4 3-52

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ATTACHMENT B

PROPOSED CHANGES TO THE TECHNICAL

SPECIFICATIONS FOR OPERATING LICENSES NPF-11 AND NPF-18

NPF-11	NPF-18
3/4 3-54	3/4 3-55
3/4 3-55	3/4 3-56
Insert G	Insert G
3/4 3-57	3/4 3-58
Insert H	Insert H
3/4 3-59	3/4 3-59
3/4 3-86	3/4 3-86
Inserts I and J	Inserts I and J
3/4 3-87	3/4 3-87
Insert K	Insert K
3/4 3-89	3/4 3-89
B 3/4 3-1	B 3/4 3-1
Inserts L and M	Inserts L and M
B 3/4 3-2	B 3/4 3-2
Inserts N and O	Inserts N and O
B 3/4 3-3	B 3/4 3-3
B 3/4 3-3a	B 3/4 3-3a
Insert P	Insert P
B 3/4 3-4	B 3/4 3-4
Inserts Q, R, and S	Inserts Q, R, and S
B 3/4 3-6	B 3/4 3-6
Insert T	Insert T

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Protection System instrumentation setpoints specified in Table 2.2.1-1 are the values at which the reactor trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

1. Intermediate Range Monitor, Neutron Flux - High

The IRM system consists of 8 chambers, 4 in each of the reactor trip systems. The IRM is a 5 decade 10 range instrument. The trip setpoint of 120 divisions of scale is active in each of the 10 ranges. Thus as the IRM is ranged up to accommodate the increase in power level, the trip setpoint is also ranged up. The IRM instruments provide for overlap with both the APRM and SRM systems.

The most significant source of reactivity changes during the power increase is due to control rod withdrawal. In order to ensure that the IRM provides the required protection, a range of rod withdrawal accidents have been analyzed. The results of these analyses are in Section 15.4.1.2 of the FSAR. The most severe case involves an initial condition in which THERMAL POWER is at approximately 1% of RATED THERMAL POWER. Additional conservatism was taken in this analysis by assuming the IRM channel closest to the control rod being withdrawn is bypassed. The results of this analysis show that the reactor is shutdown and peak power is limited to 1% of RATED THERMAL POWER with the peak fuel enthalpy well below the fuel failure threshold of 170 cal/gm. Based on this analysis, the IRM provides protection against local control rod errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

2. Average Power Range Monitor

For operation at low pressure and low flow during STARTUP, the APRM scram setting of 15% of RATED THERMAL POWER provides adequate thermal margin between the setpoint and the Safety Limits. The margin accommodates the anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor and cold water from sources available during startup is not much colder than that already in the system. Temperature coefficients are small and control rod patterns are constrained by the RSCS and RWM. Of all the possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power increase. Because

INFO ONLY - NO CHANGES

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Average Power Range Monitor (Continued)

the flux distribution associated with uniform rod withdrawals does not involve high local peaks and because several rods must be moved to change power by a significant amount, the rate of power rise is very slow. Generally the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the trip level, the rate of power rise is not more than 5% of RATED THERMAL POWER per minute and the APRM system would be more than adequate to assure shutdown before the power could exceed the Safety Limit. The 15% neutron flux trip remains active until the mode switch is placed in the Run position.

The APRM trip system is calibrated using heat balance data taken during steady state conditions. Fission chambers provide the basic input to the system and therefore the monitors respond directly and quickly to changes due to transient operation for the case of the Fixed Neutron Flux-High 118% setpoint; i.e, for a power increase, the THERMAL POWER of the fuel will be less than that indicated by the neutron flux due to the time constants of the heat transfer associated with the fuel. For the Flow Biased Simulated Thermal Power-Upscale setpoint, a time constant of 6 ± 1 seconds is introduced into the flow biased APRM in order to simulate the fuel thermal transient characteristics. A more conservative maximum value is used for the flow biased setpoint as shown in Table 2.2.1-1.

The APRM setpoints were selected to provide adequate margin for the Safety Limits and yet allow operating margin that reduces the possibility of unnecessary shutdown. The flow referenced trip setpoint must be adjusted by the specified formula in Specification 3.2.2 in order to maintain these margins when MFLPD is greater than or equal to FRTP.

3. Reactor Vessel Steam Dome Pressure-High

High pressure in the nuclear system could cause a rupture to the nuclear system process barrier resulting in the release of fission products. A pressure increase while operating will also tend to increase the power of the reactor by compressing voids thus adding reactivity. The trip will quickly reduce the neutron flux, counteracting the pressure increase. The trip setting is slightly higher than the operating pressure to permit normal operation without spurious trips. The setting provides for a wide margin to the maximum allowable design pressure and takes into account the location of the pressure measurement compared to the highest pressure that occurs in the system during a transient. This trip setpoint is effective at low power/flow conditions when the turbine stop valve closure trip is bypassed. For a turbine trip under these conditions, the transient analysis indicated an adequate margin to the thermal hydraulic limit.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2.

APPLICABILITY: As shown in Table 3.3.1-1.

INSERT "A"

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channels and/or trip system in the tripped condition* within 1 hour.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within 1 hour and take the ACTION required by Table 3.3.1-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip functional unit shown in Table 3.3.1-2 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip system.

INSERT "B"

*With a design providing only one channel per trip system, an inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.1-1 for that Trip Function shall be taken.

**If more channels are inoperable in one trip system than in the other, select that trip system to place in the tripped condition, except when this would cause the Trip Function to occur.

ATTACHMENT B

PROPOSED CHANGES TO THE TECHNICAL

SPECIFICATIONS FOR OPERATING LICENSES NPF-11 AND NPF-18

INSERT A

a. With one channel required by Table 3.3.1-1 inoperable in one or more Functional Units, place the inoperable channel and/or that trip system in the tripped condition* within 12 hours.

b. With two or more channels required by Table 3.3.1-1 inoperable in one or more Functional Units:

   1. Within one hour, verify sufficient channels remain OPERABLE or tripped* to maintain trip capability in the Functional Unit, and

   2. Within 6 hours, place the inoperable channel(s) in one trip system and/or that trip system** in the tripped condition*, and

   3. Within 12 hours, restore the inoperable channels in the other trip system to an OPERABLE status or tripped*.

c. Otherwise, take the ACTION required by Table 3.3.1-1 for the Functional Unit.

INSERT B

* An inoperable channel or trip system need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, if the inoperable channel is not restored to OPERABLE status within the required time, the ACTION required by Table 3.3.1-1 for the Functional Unit shall be taken.

** This ACTION applies to that trip system with the most inoperable channels; if both trip systems have the same number of inoperable channels, the ACTION can be applied to either trip system.

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
1. Intermediate Range Monitors:			
a. Neutron Flux - High	2	3	1
	3, 4	2	2
	5[(b)]	3	3
b. Inoperative	2	3	1
	3, 4	2	2
	5	3	3
2. Average Power Range Monitor:[(c)]			
a. Neutron Flux - High, Setdown	2	2	1
	3	2	2
	5[(b)]	2	3
b. Flow Biased Simulated Thermal Power-Upscale	1	2	4
c. Fixed Neutron Flux-High	1	2	4
d. Inoperative	1, 2	2	1
	3	2	2
	5	2	3
3. Reactor Vessel Steam Dome Pressure - High	1, 2[(d)]	2	1
4. Reactor Vessel Water Level - Low, Level 3	1, 2	2	1
5. Main Steam Line Isolation Valve - Closure	1[(e)]	4	4
6. Main Steam Line Radiation - High	1, 2[(d)]	2	5

INFO ONLY - NO CHANGES

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
7. Primary Containment Pressure - High	$1, 2^{(f)}$	$2^{(g)}$	1
8. Scram Discharge Volume Water Level - High	1, 2,	2	1
	$5^{(h)}$	2	3
9. Turbine Stop Valve - Closure	$1^{(i)}$	$4^{(j)}$	6
10. Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure - Low	$1^{(i)}$	$2^{(j)}$	6
11. Reactor Mode Switch Shutdown Position	1, 2	1	1
	3, 4	1	7
	5	1	3
12. Manual Scram	1, 2	1	1
	3, 4	1	8
	5	1	9
13. Control Rod Drive			
a. Charging Water Header Pressure - Low	2	2	1
	$5^{(h)}$	2	3
b. Delay Timer	2	2	1
	$5^{(h)}$	2	3

INFO ONLY - NO CHANGES

INFO ONLY - NO CHANGES

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION

ACTION 1 - Be in at least HOT SHUTDOWN within 12 hours.

ACTION 2 - Verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within one hour.

ACTION 3 - Suspend all operations involving CORE ALTERATIONS* and insert all insertable control rods within one hour.

ACTION 4 - Be in at least STARTUP within 6 hours.

ACTION 5 - Be in STARTUP with the main steam line isolation valves closed within 6 hours or in at least HOT SHUTDOWN within 12 hours.

ACTION 6 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure to $\leq$ 140 psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER, within 2 hours.

ACTION 7 - Verify all insertable control rods to be inserted within 1 hour.

ACTION 8 - Lock the reactor mode switch in the Shutdown position within 1 hour.

ACTION 9 - Suspend all operations involving CORE ALTERATIONS,* and insert all insertable control rods and lock the reactor mode switch in the SHUTDOWN position within 1 hour.

*Except movement of IRM, SRM or special movable detectors, or replacement of LPRM strings provided SRM instrumentation is OPERABLE per Specification 3.9.2.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to ~~2~~ 6 hours for required surveillance without placing the channel in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(b) The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn* and during shutdown margin demonstrations performed per Specification 3.10.3.

(c) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than 14 LPRM inputs to an APRM channel.

(d) This function is not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.10.1.

(e) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(f) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(g) Also actuates the standby gas treatment system.

(h) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(i) This function shall be automatically bypassed when turbine first stage pressure is $\leq$ 140 psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER.

(j) Also actuates the EOC-RPT system.

*Not required for control rods removed per Specifications 3.9.10.1 or 3.9.10.2.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High*	NA
b. Inoperative	NA
2. Average Power Range Monitor*	
a. Neutron Flux - High, Setdown	NA
b. Flow Biased Simulated Thermal Power-Upscale	$\leq$ 0.09**
c. Fixed Neutron Flux - High	$\leq$ 0.09
d. Inoperative	NA
3. Reactor Vessel Steam Dome Pressure - High	$\leq$ 0.55
4. Reactor Vessel Water Level - Low, Level 3	$\leq$ 1.05
5. Main Steam Line Isolation Valve - Closure	$\leq$ 0.06
6. Main Steam Line Radiation - High	NA
7. Primary Containment Pressure - High	NA
8. Scram Discharge Volume Water Level - High	NA
9. Turbine Stop Valve - Closure	$\leq$ 0.06
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	$\leq$ 0.08#
11. Reactor Mode Switch Shutdown Position	NA
12. Manual Scram	NA
13. Control Rod Drive	
a. Charging Water Header Pressure - Low	NA
b. Delay Timer	NA

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel.

**Not including simulated thermal power time constant.

#Measured from start of turbine control valve fast closure.

INFO ONLY - NO CHANGES.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[(a)]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors				
a. Neutron Flux - High	S/U[(b)],S S	S/U[(c)], W W	R R	2* 3*, 4, 5
b. Inoperative	NA	W	NA	2*, 3*, 4, 5
2. Average Power Range Monitor:[(f)]				
a. Neutron Flux - High, Setdown	S/U[(b)],S S	S/U[(c)], W W	SA SA	1, 2* 3*, 5
b. Flow Biased Simulated Thermal Power-Upscale	S, D[(g)]	S/U[(c)], ~~W~~ Q	W[(d)(e)], SA, R[(h)]	1
c. Fixed Neutron Flux - High	S	S/U[(c)], ~~W~~ Q	W[(d)], SA	1
d. Inoperative	NA	~~W~~ Q	NA	1, 2, 3, 5
3. Reactor Vessel Steam Dome Pressure - High	NA	~~W~~ Q	Q	1, 2
4. Reactor Vessel Water Level - Low, Level 3	~~NA~~ S	~~W~~ Q	R	1, 2
5. Main Steam Line Isolation Valve - Closure	NA	Q	R	1
6. Main Steam Line Radiation - High	S	~~W~~ Q	R	1, 2
7. Primary Containment Pressure - High	NA	~~W~~ Q	Q	1, 2

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High	NA	~~M~~ Q	R	1, 2, 5
9. Turbine Stop Valve - Closure	NA	Q	R	1
10. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	NA	Q	R	1
11. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	W	NA	1, 2, 3, 4, 5
13. Control Rod Drive				
a. Charging Water Header Pressure - Low	NA	M	R	2, 5
b. Delay Timer	NA	M	R	2, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decades during each startup and the IRM and APRM channels shall be determined to overlap for at least 1/2 decades during each controlled shutdown, if not performed within the previous 7 days.

(c) Within 24 hours prior to startup, if not performed within the previous 7 days.

(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power levels calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥ 25% of RATED THERMAL POWER. The APRM Gain Adjustment Factor (GAF) for any channel shall be equal to the power value determined by the heat balance divided by the APRM reading for that channel.

Within 2 hours, adjust any APRM channel with a GAF > 1.02. In addition, adjust any APRM channel within 12 hours, (1) if power is greater than or equal to 90% of RATED THERMAL POWER and the APRM channel GAF is < 0.98, or (2) if power is less than 90% of RATED THERMAL POWER and the APRM reading exceeds the power value determined by the heat balance by more than 10% of RATED THERMAL POWER. Until any required APRM adjustment has been accomplished, notification shall be posted on the reactor control panel.

(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH). INSERT "C"

(g) Measure and compare core flow to rated core flow.

(h) This calibration shall consist of verifying the 6 ± 1 second simulated thermal power time constant.

ATTACHMENT B

PROPOSED CHANGES TO THE TECHNICAL

SPECIFICATIONS FOR OPERATING LICENSES NPF-11 AND NPF-18

INSERT C

* The provisions of Specification 4.0.4 are not applicable for a period of 24 hours after entering OPERATIONAL CONDITION 2 or 3 when shutting down from OPERATIONAL CONDITION 1.

INSTRUMENTATION

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The isolation actuation instrumentation channels shown in Table 3.3.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.2-2 and with ISOLATION SYSTEM RESPONSE TIME as shown in Table 3.3.2-3.

APPLICABILITY: As shown in Table 3.3.2-1.

ACTION:

a. With an isolation actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel(s) and/or trip system in the tripped condition* within one hour.

c. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition*** within one hour and take the ACTION required by Table 3.3.2-1.

a

INSERT "D"

*An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.2-1 for that Trip Function shall be taken. 6

**If more channels are inoperable in one trip system than in the other, select that trip system to place in the tripped condition except when this would cause the Trip Function to occur.

***An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 1 hour or the ACTION required by Table 3.3.2-1 for that Trip Function shall be taken.

ATTACHMENT B

PROPOSED CHANGES TO THE TECHNICAL

SPECIFICATIONS FOR OPERATING LICENSES NPF-11 AND NPF-18

INSERT D

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System Requirement for one trip system, either

   1. Place the inoperable channel(s) and/or trip system in the tripped condition* within

   a) 1 hour for trip functions without an OPERABLE channel,
   b) 12 hours for trip functions common to RPS Instrumentation, and
   c) 24 hours for trip functions not common to RPS Instrumentation.

   or

   2. Take the ACTION required by Table 3.3.2-1.

c. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems,

   1. Place at least one trip system** in the tripped condition*** within one hour, and

   2. a) Place the inoperable channel(s) in the remaining trip system in the tripped condition*** within

      1) 1 hour for trip functions without an OPERABLE channel,
      2) 12 hours for trip functions common to RPS Instrumentation, and
      3) 24 hours for trip functions not common to RPS Instrumentation

      or

      b) Take the ACTION required by Table 3.3.2-1.

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TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

ACTION

within

ACTION 20 - Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN with the next 24 hours.

ACTION 21 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 22 - Close the affected system isolation valves within 1 hour and declare the affected system inoperable.

ACTION 23 - Be in at least STARTUP within 6 hours.

ACTION 24 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within 1 hour.

ACTION 25 - Lock the affected system isolation valves closed within 1 hour and declare the affected system inoperable.

ACTION 26 - Provided that the manual initiation function is OPERABLE for each other group valve, inboard or outboard, as applicable, in each line, restore the manual initiation function to OPERABLE status within 24 hours; otherwise, restore the manual initiation function to OPERABLE status within 8 hours; otherwise:

a. Be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, or

b. Close the affected system isolation valves within the next hour and declare the affected system in operable.

NOTES

* May be bypassed with reactor steam pressure ≤ 1043 psig and all turbine stop valves closed.

** When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

\# During CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

(a) See Specification 3.6.3, Table 3.6.3-1 for valves in each valve group.

(b) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the channel in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter. In addition for those trip systems with a design providing only one channel per trip system, the channel may be placed in an inoperable status for up to 8 hours for required surveillance testing without placing the channel in the tripped condition provided that the redundant isolation valve, inboard or outboard, as applicable, in each line is operable and all required actuation instrumentation for that redundant valve is OPERABLE, or place the trip system in the tripped condition.

6

(c) Also actuates the standby gas treatment system.

(d) A channel is OPERABLE if 2 of 4 instruments in that channel are OPERABLE.

(e) Also actuates secondary containment ventilation isolation dampers per Table 3.6.5.2-1.

(f) Closes only RWCU system inlet outboard valve.

TABLE 4 1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
A. AUTOMATIC INITIATION				
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1) Low, Level 3	~~NA~~ S	~~M~~ Q	R	1, 2, 3
2) Low Low, Level 2	NA	~~M~~ Q	R	1, 2, 3
3) Low Low Low, Level 1	~~NA~~ S	~~M~~ Q	R	1, 2, 3
b. Drywell Pressure - High	NA	~~M~~ Q	Q	1, 2, 3
c. Main Steam Line				
1) Radiation - High	S	~~M~~ Q	R	1, 2, 3
2) Pressure - Low	NA	~~M~~ Q	Q	1
3) Flow - High	NA	~~M~~ Q	R	1, 2, 3
d. Main Steam Line Tunnel Temperature - High	NA	~~M~~ Q	R	1, 2, 3
e. Condenser Vacuum - Low	NA	~~M~~ Q	Q	1, 2*, 3*
f. Main Steam Line Tunnel Δ Temperature - High	NA	~~M~~ Q	R	1, 2, 3
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Vent Exhaust Plenum Radiation - High	S	~~M~~ Q	R	1, 2, 3 and **
b. Drywell Pressure - High	NA	~~M~~ Q	Q	1, 2, 3
c. Reactor Vessel Water Level - Low Low, Level 2	NA	~~M~~ Q	R	1, 2, 3, and #
d. Fuel Pool Vent Exhaust Radiation - High	S	~~M~~ Q	R	1, 2, 3 and **
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	S	~~M~~ Q	R	1, 2, 3
b. Heat Exchanger Area Temperature - High	NA	~~M~~ Q	Q	1, 2, 3
c. Heat Exchanger Area Ventilation ΔT - High	NA	~~M~~ Q	Q	1, 2, 3
d. SLCS Initiation	NA	R	NA	1, 2, 3
e. Reactor Vessel Water Level - Low Low, Level 2	NA	~~M~~ Q	R	1, 2, 3

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
4. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow - High	NA	~~M~~ Q	Q	1, 2, 3
b. RCIC Steam Supply Pressure - Low	NA	~~M~~ Q	Q	1, 2, 3
c. RCIC Turbine Exhaust Diaphragm Pressure - High	NA	~~M~~ Q	Q	1, 2, 3
d. RCIC Equipment Room Temperature - High	NA	~~M~~ Q	Q	1, 2, 3
e. RCIC Steam Line Tunnel Temperature - High	NA	~~M~~ Q	Q	1, 2, 3
f. RCIC Steam Line Tunnel Δ Temperature - High	NA	~~M~~ Q	Q	1, 2, 3
g. Drywell Pressure - High	NA	~~M~~ Q	Q	1, 2, 3
h. RCIC Equipment Room Δ Temperature - High	NA	~~M~~ Q	Q	1, 2, 3
5. RHR SYSTEM STEAM CONDENSING MODE ISOLATION				
a. RHR Equipment Area Δ Temperature - High	NA	~~M~~ Q	Q	1, 2, 3
b. RHR Area Cooler Temperature - High	NA	~~M~~ Q	Q	1, 2, 3
c. RHR Heat Exchanger Steam Supply Flow - High	NA	~~M~~ Q	Q	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3	~~NA~~ S	~~M~~ Q	R	1, 2, 3
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA	~~M~~ Q	Q	1, 2, 3
c. RHR Pump Suction Flow - High	NA	~~M~~ Q	Q	1, 2, 3
d. RHR Area Temperature - High	NA	~~M~~ Q	Q	1, 2, 3
e. RHR Equipment Area ΔT - High	NA	~~M~~ Q	Q	1, 2, 3
B. MANUAL INITIATION				
1. Inboard Valves	NA	R	NA	1, 2, 3
2. Outboard Valves	NA	R	NA	1, 2, 3
3. Inboard Valves	NA	R	NA	1, 2, 3 and **,#
4. Outboard Valves	NA	R	NA	1, 2, 3 and **,#
5. Inboard Valves	NA	R	NA	1, 2, 3
6. Outboard Valves	NA	R	NA	1, 2, 3
7. Outboard Valve	NA	R	NA	1, 2, 3

*When reactor steam pressure > 1043 psig and/or any turbine stop valve is open.

**When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

#During CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

INSTRUMENTATION

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3 The emergency core cooling system (ECCS) actuation instrumentation channels shown in Table 3.3.3-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.3-2 and with EMERGENCY CORE COOLING SYSTEM RESPONSE TIME as shown in Table 3.3.3-3.

APPLICABILITY: As shown in Table 3.3.3-1.

ACTION:

a. With an ECCS actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.3-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more ECCS actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.3-1.

c. With either ADS trip system "A" or "B" inoperable, restore the inoperable trip system to OPERABLE status within:

   1. 7 days, provided that the HPCS and RCIC systems are OPERABLE.

   2. 72 hours.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 122 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.3.3.1 Each ECCS actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.3.1-1.

4.3.3.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.*

4.3.3.3 The ECCS RESPONSE TIME of each ECCS trip function shown in Table 3.3.3-3 shall be demonstrated to be within the limit at least once per 18 months. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ECCS trip system.

*The specified 18-month interval may be waived for Cycle 1 provided the surveillance is performed during Refuel 1, for LPCI A, B, and C.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION(a)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
B. DIVISION 2 TRIP SYSTEM			
1. RHR B & C (LPCI MODE)			
a. Reactor Vessel Water Level - Low, Low Low, Level 1	2(b)	1, 2, 3, 4*, 5*	30
b. Drywell Pressure - High	2(b)	1, 2, 3	30
c. LPCI B and C Injection Valve Injection Line Pressure-Low (Permissive)	1/valve	1, 2, 3 4*, 5*	32 33
d. LPCI Pump B Start Time Delay Relay	1	1, 2, 3, 4*, 5*	32
e. LPCI Pump Discharge Flow - Low (Bypass)	1/pump	1, 2, 3, 4*, 5*	31
f. Manual Initiation	1/division	1, 2, 3, 4*, 5*	34
g. LPCI B and C Injection Valve Reactor Pressure-Low (Permissive)	2	1, 2, 3 4*, 5*	30 33
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"#			
a. Reactor Vessel Water Level - Low Low Low, Level 1 coincident with	2(b)	1, 2, 3	30
b. Drywell Pressure - High	2(b)	1, 2, 3	30
c. Initiation Timer	1	1, 2, 3	32
d. Reactor Vessel Water Level - Low, Level 3 (Permissive)	1	1, 2, 3	32
e. LPCI Pump B and C Discharge Pressure - High (Permissive)	2/pump	1, 2, 3	32
f. Manual Initiation	1/division	1, 2, 3	34
g. Drywell Pressure Bypass Timer	1	1, 2, 3	32
h. Manual Inhibit	1/division	1, 2, 3	34

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION[a]	APPLICABLE OPERATIONAL CONDITIONS	ACTION
C. DIVISION 3 TRIP SYSTEM			
1. HPCS SYSTEM			
a. Reactor Vessel Water Level - Low, Low, Level 2	4[b]	1, 2, 3, 4*, 5*	35
b. Drywell Pressure - High	4[b]	1, 2, 3	35
c. Reactor Vessel Water Level-High, Level 8	2[c]	1, 2, 3, 4*, 5*	32
d. Deleted			
e. Deleted			
f. Pump Discharge Pressure-High (Bypass)	1	1, 2, 3, 4*, 5*	31
g. HPCS System Flow Rate-Low (Permissive)	1	1, 2, 3, 4*, 5*	31
h. Manual Initiation	1/division	1, 2, 3, 4*, 5*	34

D. LOSS OF POWER	TOTAL NO. OF INSTRUMENTS	INSTRUMENTS TO TRIP	MINIMUM OPERABLE INSTRUMENTS[d]	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	2/bus	2/bus	2/bus	1, 2, 3, 4**, 5**	37
2. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	2/bus	2/bus	2/bus	1, 2, 3, 4**, 5**	37

(a) A channel instrument may be placed in an inoperable status for up to ~~2~~ 6 hours during periods of required surveillance without placing the trip system/channel/instrument in the tripped condition provided at least one other OPERABLE channel/instrument in the same trip system is monitoring that parameter.

(b) Also actuates the associated division diesel generator.

(c) Provides signal to close HPCS pump discharge valve only on 2-out-of-2 logic.

* Applicable when the system is required to be OPERABLE per Specification 3.5.2 or 3.5.3.

** Required when ESF equipment is required to be OPERABLE.

\# Not required to be OPERABLE when reactor steam dome pressure is $\leq$ 122 psig.

INSERT "E"

ATTACHMENT B

PROPOSED CHANGES TO THE TECHNICAL

SPECIFICATIONS FOR OPERATING LICENSES NPF-11 AND NPF-18

INSERT E

(d) A channel/instrument may be placed in an inoperable status for up to 2 hours during periods of required surveillance without placing the trip system/channel/instrument in the tripped condition provided at least one other OPERABLE channel/instrument in the same trip system is monitoring that parameter.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION
ACTION

ACTION 30 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

a. With one channel inoperable, place the inoperable channel in the tripped condition within one hour or declare the associated system inoperable. 24 hours

b. With more than one channel inoperable, declare the associated system inoperable.

ACTION 31 - With the number of OPERABLE channels less than required by the Minimum OPERABLE channels per Trip Function requirement; place the a inoperable channel in the tripped condition within one hour 24 hours; restore the inoperable channel to OPERABLE status within 7 days or declare the associated system inoperable.

ACTION 32 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, declare the associated ADS trip system or ECCS inoperable within 24 hours.

ACTION 33 - With the number of OPERABLE channels less than the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour 24 hours.

ACTION 34 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 8 24 hours or declare the associated ADS trip system or ECCS inoperable.

ACTION 35 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement

a. For one trip system, place that trip system in the tripped condition within one hour or declare the HPCS system inoperable. 24 hours

b. For both trip systems, declare the HPCS system inoperable.

ACTION 36 - Deleted

ACTION 37 - With the number of OPERABLE instruments less than the Minimum Operable Instruments, place the inoperable instrument(s) in the tripped condition within 1 hour or declare the associated emergency diesel generator inoperable and take the ACTION required by Specification 3.8.1.1 or 3.8.1.2 as appropriate.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION

ACTION 38 With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per trip function requirements:

a. With one channel inoperable, remove the inoperable channel within one hour; restore the inoperable channel to OPERABLE status within 7 days or declare the associated ECCS systems inoperable.

24 hours

b. With both channels inoperable, restore at least one channel to OPERABLE status within one hour or declare the associated ECCS systems inoperable.

TABLE 4.3.3.1-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
A. DIVISION I TRIP SYSTEM				
1. RHR-A (LPCI MODE) AND LPCS SYSTEM				
a. Reactor Vessel Water Level - Low Low Low, Level 1	~~NA~~ S	~~M~~ Q	R	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	NA	~~M~~ Q	Q	1, 2, 3
c. LPCS Pump Discharge Flow-Low	NA	~~M~~ Q	Q	1, 2, 3, 4*, 5*
d. LPCS and LPCI A Injection Valve Injection Line Pressure Low Interlock	NA	~~M~~ Q	R	1, 2, 3, 4*, 5*
e. LPCS and LPCI A Injection Valve Reactor Pressure Low Interlock	NA	~~M~~ Q	R	1, 2, 3, 4*, 5*
f. LPCI Pump A Start Time Delay Relay	NA	~~M~~ Q	Q	1, 2, 3, 4*, 5*
g. LPCI Pump A Flow-Low	NA	~~M~~ Q	Q	1, 2, 3, 4*, 5*
h. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"#				
a. Reactor Vessel Water Level - Low Low Low, Level 1	~~NA~~ S	~~M~~ Q	R	1, 2, 3
b. Drywell Pressure-High	NA	~~M~~ Q	Q	1, 2, 3
c. Initiation Timer	NA	~~M~~ Q	Q	1, 2, 3
d. Reactor Vessel Water Level - Low, Level 3	~~NA~~ S	~~M~~ Q	R	1, 2, 3
e. LPCS Pump Discharge Pressure-High	NA	~~M~~ Q	Q	1, 2, 3
f. LPCI Pump A Discharge Pressure-High	NA	~~M~~ Q	Q	1, 2, 3
g. Manual Initiation	NA	R	NA	1, 2, 3
h. Drywell Pressure Bypass Timer	NA	~~M~~ Q	Q	1, 2, 3
i. Manual Inhibit	NA	R	NA	1, 2, 3

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
B. DIVISION 2 TRIP SYSTEM				
1. RHR B AND C (LPCI MODE)				
a. Reactor Vessel Water Level - Low Low Low, Level 1	~~NA~~ S	~~M~~ Q	R	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	NA	~~M~~ Q	Q	1, 2, 3
c. LPCI B and C Injection Valve Injection Line Pressure Low Interlock	NA	~~M~~ Q	R	1, 2, 3, 4*, 5*
d. LPCI Pump B Start Time Delay Relay	NA	~~M~~ Q	Q	1, 2, 3, 4*, 5*
e. LPCI Pump Discharge Flow-Low	NA	~~M~~ Q	Q	1, 2, 3, 4*, 5*
f. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
g. LPCI B and C Injection Valve Reactor Pressure Low Interlock	NA	~~M~~ Q	R	1, 2, 3, 4*, 5*
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"#				
a. Reactor Vessel Water Level - Low Low Low, Level 1	~~NA~~ S	~~M~~ Q	R	1, 2, 3
b. Drywell Pressure-High	NA	~~M~~ Q	Q	1, 2, 3
c. Initiation Timer	NA	~~M~~ Q	Q	1, 2, 3
d. Reactor Vessel Water Level - Low, Level 3	~~NA~~ S	~~M~~ Q	R	1, 2, 3
e. LPCS Pump B and C Discharge Pressure-High	NA	~~M~~ Q	Q	1, 2, 3
f. Manual Initiation	NA	R	NA	1, 2, 3
h. Drywell Pressure Bypass Timer	NA	~~M~~ Q	Q	1, 2, 3
i. Manual Inhibit	NA	R	NA	1, 2, 3

LPCI

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
C. DIVISION 3 TRIP SYSTEM				
1. HPCS SYSTEM				
a. Reactor Vessel Water Level - Low Low, Level 2	~~NA~~ S	~~M~~ Q	R	1, 2, 3, 4*, 5*
b. Drywell Pressure-High	NA	~~M~~ Q	Q	1, 2, 3
c. Reactor Vessel Water Level-High Level 8	~~NA~~ S	~~M~~ Q	R	1, 2, 3, 4*, 5*
d. Deleted				
e. Deleted				
f. Pump Discharge Pressure-High	NA	~~M~~ Q	Q	1, 2, 3, 4*, 5*
g. HPCS System Flow Rate-Low	NA	~~M~~ Q	Q	1, 2, 3, 4*, 5*
h. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
D. LOSS OF POWER				
1. 4.16 kV Emergency Bus Under-voltage (Loss of Voltage)	NA	NA	R	1, 2, 3, 4**, 5**
2. 4.16 kV Emergency Bus Under-voltage (Degraded Voltage)	NA	NA	R	1, 2, 3, 4**, 5**

#Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 122 psig.
*When the system is required to be OPERABLE after being manually realigned, as applicable, per Specification 3.5.2.
**Required when ESF equipment is required to be OPERABLE.

INSTRUMENTATION

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.1 The anticipated transient without scram recirculation pump trip (ATWS-RPT) system instrumentation channels shown in Table 3.3.4.1-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4.1-2.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION:

a. With an ATWS recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.1-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within ~~1 hour~~ 24 hours

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system and:

   1. If the inoperable channels consist of one reactor vessel water level channel and one reactor vessel pressure channel, place both inoperable channels in the tripped condition within ~~1 hour~~ 24 hours, or, if this action will initiate a pump trip, declare the trip system inoperable.

   2. If the inoperable channels include two reactor vessel water level channels or two reactor vessel pressure channels, declare the trip system inoperable.

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within 1 hour or be in at least STARTUP within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.3.4.1.1 Each ATWS recirculation pump trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.4.1-1.

4.3.4.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

TABLE 3.3.4.1-1

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM [(a)]
1. Reactor Vessel Water Level - Low Low, Level 2	2
2. Reactor Vessel Pressure-High	2

[(a)] One channel in one trip system may be placed in an inoperable status for up to ~~2~~ 6 hours for required surveillance provided that all other channels are OPERABLE.

TABLE 3.3.4.1-2

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel, Water Level - Low Low, Level 2	$\geq$- 50 inches*	$\geq$- 57 inches*
2. Reactor Vessel Pressure-High	$\leq$ 1135 psig	$\leq$ 1150 psig

* See Bases Figure B3/4 3-1.

No changes
Info Only

TABLE 4.3.4.1-1

ATWS RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Reactor Vessel Water Level - Low Low, Level 2	S	~~M~~ Q	R
2. Reactor Vessel Pressure - High	S	~~M~~ Q	Q

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.2 The end-of-cycle recirculation pump trip (EOC-RPT) system instrumentation channels shown in Table 3.3.4.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4.2-2 and with the END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME as shown in Table 3.3.4.2-3.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER.

ACTION:

a. With an end-of-cycle recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within 1 hour. 12 hours

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement(s) for one trip system and:

   1. If the inoperable channels consist of one turbine control valve channel and one turbine stop valve channel, place both inoperable channels in the tripped condition within 1 hour. 12 hours

   2. If the inoperable channels include two turbine control valve channels or two turbine stop valve channels, declare the trip system inoperable.

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours. Otherwise, either:

   1. Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Limiting Condition for Operation (LCO) to the EOC-RPT inoperable value per Specification 3.2.3 within the next 1 hour or,

   2. Reduce THERMAL POWER to less than 30% of RATED THERMAL POWER within the next 6 hours.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within 1 hour. Otherwise, either:

INFO ONLY - NO CHANGES

INSTRUMENTATION

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

1. Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Limiting Condition for Operation (LCO) to the EOC-RPT inoperable value per Specification 3.2.3 within the next 1 hour or,

2. reduce THERMAL POWER to less than 30% of RATED THERMAL POWER within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.3.4.2.1 Each end-of-cycle recirculation pump trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.4.2.1-1.

4.3.4.2.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.4.2.3 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME of each trip function shown in Table 3.3.4.2-3 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least the logic of one type of channel input, turbine control valve fast closure or turbine stop valve closure, such that both types of channel inputs are tested at least once per 36 months. The time allotted for breaker arc suppression shall be verified by test at least once per 60 months.

TABLE 3.3.4.2-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[(a)]
1. Turbine Stop Valve - Closure	2[(b)]
2. Turbine Control Valve - Fast Closure	2[(b)]

[(a)]A trip system may be placed in an inoperable status for up to ~~2~~ 6 hours for required surveillance provided that the other trip system is OPERABLE.

[(b)]This function shall be automatically bypassed when turbine first stage pressure is less than or equal to 140 psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER.

TABLE 4.3.4.2.1-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Turbine Stop Valve-Closure	Q	R
2. Turbine Control Valve-Fast Closure	Q	R

INFO ONLY - NO CHANGES .

INFO ONLY - NO CHANGES

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.5 The reactor core isolation cooling (RCIC) system actuation instrumentation channels shown in Table 3.3.5-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.5-2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3 with reactor steam dome pressure greater than 150 psig.

ACTION:

a. With a RCIC system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.5-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more RCIC system actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.5-1.

SURVEILLANCE REQUIREMENTS

4.3.5.1 Each RCIC system actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.5.1-1.

4.3.5.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

TABLE 3.3.5-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

FUNCTIONAL UNITS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[(a)]	ACTION
a. Reactor Vessel Water Level - Low Low, Level 2	2	50
b. Reactor Vessel Water Level - High, Level 8	2[(b)]	51
c. Manual Initiation	1[(c)]	52

(a) A channel may be placed in an inoperable status for up to ~~2~~ 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

(b) One trip system with two-out-of-two logic.

(c) Single channel.

TABLE 3.3.5-1 (Continued)

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION 50 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement:

a. For one trip system, place the inoperable channel in the tripped condition within one hour or declare the RCIC system inoperable.
24 hours

b. For both trip systems, declare the RCIC system inoperable.

ACTION 51 - With the number of OPERABLE channel less than required by the minimum OPERABLE Channels per Trip System requirement, declare the RCIC system inoperable. within 24 hours

ACTION 52 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, restore the inoperable channel to OPERABLE status within 8 hours or declare the RCIC system inoperable.
24

TABLE 4.3.5.1-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNITS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
a. Reactor Vessel Water Level - Low Low, Level 2	NA	~~M~~ Q	R
b. Reactor Vessel Water Level - High, Level 8	~~NA~~ S	~~M~~ Q	R
c. Manual Initiation	NA	R	NA

3/4.3.6 CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6 The control rod withdrawal block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

a. With a control rod withdrawal block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, take the ACTION required by Table 3.3.6-1.

SURVEILLANCE REQUIREMENTS

4.3.6 Each of the above required control rod withdrawal block trip systems and instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.6-1. (*)

INSERT "F"

ATTACHMENT B

PROPOSED CHANGES TO THE TECHNICAL

SPECIFICATIONS FOR OPERATING LICENSES NPF-11 AND NPF-18

INSERT F

* A channel may be placed in an inoperable status for up to 6 hours for required surveillance (or 12 hours for repair) without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

TABLE 3.3.6-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

ACTION

ACTION 60 - Declare the RBM inoperable and take the ACTION required by Specification 3.1.4.3.

ACTION 61 - With the number of OPERABLE channels:

a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 7 days or place the inoperable channel in the tripped condition within the next hour.

b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.

ACTION 62 - With the number of OPERABLE Channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour.

12 hours

NOTE

* With THERMAL POWER ≥ 30% of RATED THERMAL POWER.

** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

a. The RBM shall be automatically bypassed when a peripheral control rod is selected.

b. This function shall be automatically bypassed if detector count rate is ≥ 100 cps or the IRM channels are on range 3 or higher.

c. This function shall be automatically bypassed when the associated IRM channels are on range 8 or higher.

d. This function shall be automatically bypassed when the IRM channels are on range 3 or higher.

e. This function shall be automatically bypassed when the IRM channels are on range 1.

TABLE 4.3.6-1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	NA	S/U(b)(c), ~~M~~(c) Q	Q	1*
b. Inoperative	NA	S/U(b)(c), ~~M~~(c) Q	N.A.	1*
c. Downscale	NA	S/U(b)(c), ~~M~~(c) Q	Q	1*
2. APRM				
a. Flow Biased Simulated Thermal Power-Upscale	NA	S/U(b), ~~M~~ Q	SA	1
b. Inoperative	NA	S/U(b), ~~M~~ Q	N.A.	1, 2, 5
c. Downscale	NA	S/U(b), ~~M~~ Q	SA	1
d. Neutron Flux-High	NA	S/U(b), ~~M~~ Q	SA	2, 5
3. SOURCE RANGE MONITORS				***
a. Detector not full in	NA	S/U(b), W	N.A.	2, 5
b. Upscale	NA	S/U(b), W	Q	2, 5
c. Inoperative	NA	S/U(b), W	N.A.	2, 5
d. Downscale	NA	S/U(b), W	Q	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	S/U(b), W	N.A.	2, 5
b. Upscale	NA	S/U(b), W	Q	2, 5
c. Inoperative	NA	S/U(b), W	N.A.	2, 5
d. Downscale	NA	S/U(b), W	Q	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	NA	Q	R	1, 2, 5**
b. Scram Discharge Volume Switch in Bypass	NA	~~M~~ Q	N.A.	5**
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	NA	S/U(b), ~~M~~ Q	Q	1
b. Inoperative	NA	S/U(b), ~~M~~ Q	N.A.	1
c. Comparator	NA	S/U(b), ~~M~~ Q	Q	1

TABLE 4.3.6-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES:

a. Neutron detectors may be excluded from CHANNEL CALIBRATION.

b. Within 24 hours prior to startup, if not performed within the previous 7 days.

c. Includes reactor manual control multiplexing system input.

* With THERMAL POWER $\geq$ 30% of RATED THERMAL POWER.

** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

INSERT "G"

ATTACHMENT B

PROPOSED CHANGES TO THE TECHNICAL

SPECIFICATIONS FOR OPERATING LICENSES NPF-11 AND NPF-18

INSERT G

*** The provisions of Specification 4.0.4 are not applicable for a period of 24 hours after entering OPERATIONAL CONDITION 2 or 3 when shutting down from OPERATIONAL CONDITION 1.

INFO ONLY - NO CHANGES

3/4.3.7 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.1 The radiation monitoring instrumentation channels shown in Table 3.3.7.1-1 shall be OPERABLE* with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3.7.1-1.

ACTION:

a. With a radiation monitoring instrumentation channel alarm/trip setpoint exceeding the value shown in Table 3.3.7.1-1, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION required by Table 3.3.7.1-1.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.1 Each of the above required radiation monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the conditions and at the frequencies shown in Table 4.3.7.1-1.

---

*The normal or emergency power source may be inoperable in OPERATIONAL CONDITION 4 or 5 or when defueled.

TABLE 3.3.7.1-1

RADIATION MONITORING INSTRUMENTATION

INSTRUMENTATION	MINIMUM CHANNELS OPERABLE	APPLICABLE CONDITIONS	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
a. Main Control Room Atmospheric Control System Radiation Monitoring Subsystem	2/intake**	1,2,3,5 and *	3.5 mR/hr	0.1 to 10,000 mR/hr	70

NOTES

*When irradiated fuel is being handled in the secondary containment.

INSERT "H"

ATTACHMENT B

PROPOSED CHANGES TO THE TECHNICAL

SPECIFICATIONS FOR OPERATING LICENSES NPF-11 AND NPF-18

INSERT H

** A channel may be placed in an inoperable status for up to 6 hours for required surveillance testing without placing the Trip System in the tripped condition, provided at least one other operable channel in the same Trip System is monitoring that Trip Function.

INFO ONLY - NO CHANGES

TABLE 3.3.7.1-1 (Continued)

RADIATION MONITORING INSTRUMENTATION

ACTION

ACTION 70 -

a. With one of the required monitors inoperable, place the inoperable channel in the downscale tripped condition within 1 hour; restore the inoperable channel to OPERABLE status within 7 days, or, within the next 6 hours, initiate and maintain operation of the control room emergency filtration system in the pressurization mode of operation.

b. With both of the required monitors inoperable, initiate and maintain operation of the control room emergency filtration system in the pressurization mode of operation within 1 hour.

TABLE 4.3.7.1-1

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTATION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
a. Main Control Room Atmospheric Control System Radiation Monitoring Subsystem	S	Q	R	1,2,3,5 and *

NOTES

*When irradiated fuel is being handled in the secondary containment.

3/4.3.8 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.8 The feedwater/main turbine trip system actuation instrumentation channels shown in Table 3.3.8-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.8-2.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION:

INSERT "I"

a. With a feedwater/main turbine trip system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.8-2, declare the channel inoperable and either place the inoperable channel in the tripped condition until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value, or declare the associated system inoperable.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement, restore the inoperable channel to OPERABLE status within 7 days or be in at least STARTUP within the next 6 hours.

c. With the number of OPERABLE channels two less than required by the Minimum OPERABLE Channels per Trip System requirement, restore at least one of the inoperable channels to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.3.8.1 Each feedwater/main turbine trip system actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.8.1-1.

4.3.8.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.*

INSERT "J"

*The specified 18 month interval may be waived for Cycle 1 provided the surveillance is performed during Refuel 1.

ATTACHMENT B

PROPOSED CHANGES TO THE TECHNICAL

SPECIFICATIONS FOR OPERATING LICENSES NPF-11 AND NPF-18

INSERT I

a. With a feedwater/main turbine trip system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.8-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than that required by the Minimum OPERABLE Channels per Trip System requirement:

   1. Within 7 days, either place the inoperable channel in the tripped* condition or restore the inoperable channel to OPERABLE status.

   2. Otherwise, be in at least STARTUP within 6 hours.

c. With the number of OPERABLE channels two less than required by the Minimum OPERABLE Channels per Trip System requirement:

   1. Within two hours place or verify at least one inoperable channel in the tripped* condition, and restore either inoperable channel to OPERABLE status within 72 hours, or,

   2. Be in at least STARTUP within the next 6 hours.

INSERT J

* An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur.

TABLE 3.3.8-1

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM
a. Reactor Vessel Water Level-High, Level 8	3#

INSERT "K"

ATTACHMENT B

PROPOSED CHANGES TO THE TECHNICAL

SPECIFICATIONS FOR OPERATING LICENSES NPF-11 AND NPF-18

INSERT K

* A channel may be placed in an inoperable status for up to 6 hours for required surveillance testing without placing the Trip System in the tripped condition.

TABLE 3.3.8-2

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
a. Reactor Vessel Water Level-High, Level 8	< 55.5 inches*	< 56.0 inches*

*See Bases Figure B 3/4 3-1.

INFO ONLY-NO CHANGES

TABLE 4.3.8.1-1

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
a. Reactor Vessel Water Level-High, Level 8	S	~~M~~ Q	R

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

a. Preserve the integrity of the fuel cladding.

b. Preserve the integrity of the reactor coolant system.

c. Minimize the energy which must be adsorbed following a loss-of-coolant accident, and

d. Prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. The system meets the intent of IEEE-279, 1971, for nuclear power plant protection systems. Specified surveillance intervals for MSIV-Closure, TSV-Closure, TCV-Closure, and the Manual Scram have been determined in accordance with NEDC-30851P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System", March 1988. The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1. INSERT "M"

The measurement of response time at the specified frequencies provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the accident analysis. No credit was taken for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurement, provided such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) inplace, onsite or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

and surveillance and maintenance outage times

INSERT "L."

ATTACHMENT B

PROPOSED CHANGES TO THE TECHNICAL

SPECIFICATIONS FOR OPERATING LICENSES NPF-11 AND NPF-18

INSERT L

and MDE-83-0485 Revision 3, "Technical Specification Improvement Analysis for the Reactor Protection System for LaSalle County Station, Units 1 and 2", April 1991.

INSERT M

When a channel is placed in an inoperable status solely for performance of required surveillances, entry into LCO and required ACTIONS may be delayed, provided the associated function maintains RPS trip capability.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

INSERT "N"

~~This specification ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents by prescribing the OPERABILITY trip setpoints and response times for isolation of the reactor systems. When necessary, one channel may be inoperable for brief intervals to conduct required surveillance. Both channels of each trip system for the main steam tunnel ambient temperature and ventilation system differential temperature may be placed in an inoperable status for up to 4 hours for required reactor building ventilation system maintenance and testing and 12 hours for the required secondary containment Leak Rate test without placing the trip system in the tripped condition. This will allow for maintaining the reliability of the ventilation system and secondary containment.~~ Some of the trip settings may have tolerances explicitly stated where both the high and low values are critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting have a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

Except for the MSIVs, the safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. For A.C. operated valves, it is assumed that the A.C. power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of 13 seconds is assumed before the valve starts to move. The safety analysis considers an allowable inventory loss which in turn determines the valve speed in conjunction with the 13 second delay.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

The emergency core cooling system actuation instrumentation is provided to initiate actions to mitigate the consequences of accidents that are beyond the ability of the operator to control. This specification provides the OPERABILITY requirements, trip setpoints and response times that will ensure effectiveness of the systems to provide the design protection. Although the instruments are listed by system, in some cases the same instrument may be used to send the actuation signal to more than one system at the same time.

INSERT "O"

ATTACHMENT B

PROPOSED CHANGES TO THE TECHNICAL

SPECIFICATIONS FOR OPERATING LICENSES NPF-11 AND NPF-18

INSERT N

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P-A, Supplement 2, "Technical Specification Improvement Analyses for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation", March 1989, and with NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation", July 1990. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into LCO and required ACTIONS may be delayed, provided the associated function maintains primary containment isolation capability.

INSERT O

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30936P-A, "Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation)", Parts 1 and 2, December 1988, and RE-025 Revision 1, "Technical Specification Improvement Analysis for the Emergency Core Cooling System Actuation Instrumentation for LaSalle County Station, Units 1 and 2", April 1991. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into LCO and required ACTIONS may be delayed, provided the associated function maintains ECCS initiation capability.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971 and NEDO-24222, dated December, 1979, and Appendix G of the FSAR.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a part of the Reactor Protection System and is an essential safety supplement to the reactor trip. The purpose of the EOC-RPT is to recover the loss of thermal margin which occurs at the end-of-cycle. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity to the reactor system at a faster rate than the control rods add negative scram reactivity. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

A generic analysis, which provides for continued operation with one or both trip systems of the EOC-RPT system inoperable, has been performed. The analysis determined bounding cycle independent MINIMUM CRITICAL POWER RATIO (MCPR) Limiting Condition for Operation (LCO) values which must be used if the EOC-RPT system is inoperable. These values ensure that adequate reactivity margin to the MCPR safety limit exists in the event of the analyzed transient with the RPT function inoperable. The analysis results are further discussed in the bases for Specification 3.2.3.

A fast closure sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a position switch for each of two turbine stop valves provides input to one EOC-RPT system; a position switch from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 30% of RATED THERMAL POWER are annunciated in the control room.

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with the following:

1. NEDC-30851P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System", March 1988.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

2. GENE-770-06-1-A, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications", December 1992.

The EOC-RPT system response time is the time assumed in the analysis between initiation of valve motion and complete suppression of the electric arc, i.e., 190 ms, less the time allotted for sensor response, i.e., 10 ms, and less the time allotted for breaker arc suppression determined by test, as correlated to manufacturer's test results, i.e., 83 ms, and plant pre-operational test results.

INSERT "P"

ATTACHMENT B

PROPOSED CHANGES TO THE TECHNICAL

SPECIFICATIONS FOR OPERATING LICENSES NPF-11 AND NPF-18

INSERT P

When a channel is placed in an inoperable status solely for performance of required surveillances, entry into LCO and required ACTIONS may be delayed, provided the associated function maintains the applicable RPT initiation capability.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without providing actuation of any of the emergency core cooling equipment.

INSERT "Q"

3/4.3.6 CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of the specifications in Section 3/4.1.4, Control Rod Program Controls and Section 3/4.2 Power Distribution Limits. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

INSERT "R"

3/4.3.7 MONITORING INSTRUMENTATION

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring instrumentation ensures that; (1) the radiation levels are continually measured in the areas served by the individual channels, and (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

INSERT "S"

3.4.3.7.2 SEISMIC MONITORING INSTRUMENTATION

The OPERABILITY of the seismic monitoring instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the unit. This instrumentation is consistent with the recommendations of Regulatory Guide 1.12 "Instrumentation for Earthquakes", April 1974.

3/4.3.7.3 METEOROLOGICAL MONITORING INSTRUMENTATION

The OPERABILITY of the meteorological monitoring instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public. This instrumentation is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," February, 1972.

3/4.3.7.4 REMOTE SHUTDOWN MONITORING INSTRUMENTATION

The OPERABILITY of the remote shutdown monitoring instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN of the unit from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

ATTACHMENT B

PROPOSED CHANGES TO THE TECHNICAL

SPECIFICATIONS FOR OPERATING LICENSES NPF-11 AND NPF-18

INSERT Q

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with GENE-770-06-2-A, "Addendum To Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications (BWR RCIC Instrumentation)", December 1992. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into LCO and required ACTIONS may be delayed, provided the associated function maintains RCIC initiation capability.

INSERT R

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P-A, Supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation", October 1988, and GENE-770-06-1-A, "Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times for Selected Instrumentation Technical Specifications", December 1992. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into LCO and required ACTIONS may be delayed, provided the associated function maintains Control Rod Block capability.

INSERT S

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with GENE-770-06-1-A, "Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times for Selected Instrumentation Technical Specifications", December 1992. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into LCO and required ACTIONS may be delayed, provided the associated function maintains initiation capability.

3/4.3.7.10 DELETED

3/4.3.7.11 EXPLOSIVE GAS MONITORING INSTRUMENTATION

This instrumentation provides for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the waste gas holdup system.

3/4.3.7.12 LOOSE-PART DETECTION SYSTEM

The OPERABILITY of the loose-part detection system ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors."

3/4.3.8 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

The feedwater/main turbine trip system actuation instrumentation is provided to initiate the feedwater system/main turbine trip system in the event of reactor vessel water level equal to or greater than the level 8 setpoint associated with a feedwater controller failure, to prevent overfilling the reactor vessel which may result in high pressure liquid discharge through the safety/relief valve discharge lines.

INSERT "T"

ATTACHMENT B

PROPOSED CHANGES TO THE TECHNICAL

SPECIFICATIONS FOR OPERATING LICENSES NPF-11 AND NPF-18

INSERT T

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with GENE-770-06-1-A, "Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times for Selected Instrumentation Technical Specifications", December 1992. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into LCO and required ACTIONS may be delayed, provided the associated function maintains Feedwater System/Main Turbine Trip System actuation capability.

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Protection System instrumentation setpoints specified in Table 2.2.1-1 are the values at which the reactor trips are set for each parameter. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

1. Intermediate Range Monitor, Neutron Flux - High

The IRM system consists of 8 chambers, 4 in each of the reactor trip systems. The IRM is a 5 decade 10 range instrument. The trip setpoint of 120 divisions of scale is active in each of the 10 ranges. Thus as the IRM is ranged up to accommodate the increase in power level, the trip setpoint is also ranged up. The IRM instruments provide for overlap with both the APRM and SRM systems.

The most significant source of reactivity changes during the power increase is due to control rod withdrawal. In order to ensure that the IRM provides the required protection, a range of rod withdrawal accidents have been analyzed. The results of these analyses are in Section 15.4.1.2 of the FSAR. The most severe case involves an initial condition in which THERMAL POWER is at approximately 1% of RATED THERMAL POWER. Additional conservatism was taken in this analysis by assuming the IRM channel closest to the control rod being withdrawn is bypassed. The results of this analysis show that the reactor is shutdown and peak power is limited to 1% of RATED THERMAL POWER with the peak fuel enthalpy well below the fuel failure threshold of 170 cal/gm. Based on this analysis, the IRM provides protection against local control rod errors and continuous withdrawal of control rods in sequence and provides backup protection for the APRM.

2. Average Power Range Monitor

For operation at low pressure and low flow during STARTUP, the APRM scram setting of 15% of RATED THERMAL POWER provides adequate thermal margin between the setpoint and the Safety Limits. The margin accommodates the anticipated maneuvers associated with power plant startup. Effects of increasing pressure at zero or low void content are minor and cold water from sources available during startup is not much colder than that already in the system. Temperature coefficients are small and control rod patterns are constrained by the RSCS and RWM. Of all the possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power increase. Because

INFO ONLY - NO CHANGES

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Average Power Range Monitor (Continued)

the flux distribution associated with uniform rod withdrawals does not involve high local peaks and because several rods must be moved to change power by a significant amount, the rate of power rise is very slow. Generally the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the trip level, the rate of power rise is not more than 5% of RATED THERMAL POWER per minute and the APRM system would be more than adequate to assure shutdown before the power could exceed the Safety Limit. The 15% neutron flux trip remains active until the mode switch is placed in the Run position.

The APRM trip system is calibrated using heat balance data taken during steady state conditions. Fission chambers provide the basic input to the system and therefore the monitors respond directly and quickly to changes due to transient operation for the case of the Fixed Neutron Flux-High 118% setpoint; i.e, for a power increase, the THERMAL POWER of the fuel will be less than that indicated by the neutron flux due to the time constants of the heat transfer associated with the fuel. For the Flow Biased Simulated Thermal Power-Upscale setpoint, a time constant of 6 ± 1 seconds is introduced into the flow biased APRM in order to simulate the fuel thermal transient characteristics. A more conservative maximum value is used for the flow biased setpoint as shown in Table 2.2.1-1.

The APRM setpoints were selected to provide adequate margin for the Safety Limits and yet allow operating margin that reduces the possibility of unnecessary shutdown. The flow referenced trip setpoint must be adjusted by the specified formula in Specification 3.2.2 in order to maintain these margins when MFLPD is greater than or equal to FRTP.

3. Reactor Vessel Steam Dome Pressure-High

High pressure in the nuclear system could cause a rupture to the nuclear system process barrier resulting in the release of fission products. A pressure increase while operating will also tend to increase the power of the reactor by compressing voids thus adding reactivity. The trip will quickly reduce the neutron flux, counteracting the pressure increase. The trip setting is slightly higher than the operating pressure to permit normal operation without spurious trips. The setting provides for a wide margin to the maximum allowable design pressure and takes into account the location of the pressure measurement compared to the highest pressure that occurs in the system during a transient. This trip setpoint is effective at low power/flow conditions when the turbine stop valve closure trip is bypassed. For a turbine trip under these conditions, the transient analysis indicated an adequate margin to the thermal hydraulic limit.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2.

APPLICABILITY: As shown in Table 3.3.1-1.

INSERT "A"

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channels and/or trip system in the tripped condition* within 1 hour.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within 1 hour and take the ACTION required by Table 3.3.1-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip functional unit shown in Table 3.3.1-2 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip system.

INSERT "B"

*With a design providing only one channel per trip system, an inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.1-1 for that Trip Function shall be taken.

**If more channels are inoperable in one trip system than in the other, select that trip system to place in the tripped condition, except when this would cause the Trip Function to occur.

ATTACHMENT B

PROPOSED CHANGES TO THE TECHNICAL

SPECIFICATIONS FOR OPERATING LICENSES NPF-11 AND NPF-18

INSERT A

a. With one channel required by Table 3.3.1-1 inoperable in one or more Functional Units, place the inoperable channel and/or that trip system in the tripped condition* within 12 hours.

b. With two or more channels required by Table 3.3.1-1 inoperable in one or more Functional Units:

   1. Within one hour, verify sufficient channels remain OPERABLE or tripped* to maintain trip capability in the Functional Unit, and

   2. Within 6 hours, place the inoperable channel(s) in one trip system and/or that trip system** in the tripped condition*, and

   3. Within 12 hours, restore the inoperable channels in the other trip system to an OPERABLE status or tripped*.

c. Otherwise, take the ACTION required by Table 3.3.1-1 for the Functional Unit.

INSERT B

* An inoperable channel or trip system need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, if the inoperable channel is not restored to OPERABLE status within the required time, the ACTION required by Table 3.3.1-1 for the Functional Unit shall be taken.

** This ACTION applies to that trip system with the most inoperable channels; if both trip systems have the same number of inoperable channels, the ACTION can be applied to either trip system.

INFO ONLY - NO CHANGES.

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
1. Intermediate Range Monitors:			
a. Neutron Flux - High	2	3	1
	3, 4	2	2
	$5^{(b)}$	3	3
b. Inoperative	2	3	1
	3, 4	2	2
	5	3	3
2. Average Power Range Monitor:(c)			
a. Neutron Flux - High, Setdown	2	2	1
	3	2	2
	$5^{(b)}$	2	3
b. Flow Biased Simulated Thermal Power-Upscale	1	2	4
c. Fixed Neutron Flux-High	1	2	4
d. Inoperative	1, 2	2	1
	3	2	2
	5	2	3
3. Reactor Vessel Steam Dome Pressure - High	$1, 2^{(d)}$	2	1
4. Reactor Vessel Water Level - Low, Level 3	1, 2	2	1
5. Main Steam Line Isolation Valve - Closure	$1^{(e)}$	4	4
6. Main Steam Line Radiation - High	$1, 2^{(d)}$	2	5

TABLE 3.3.1-1 (Continued)
REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
7. Primary Containment Pressure - High	1, $2^{(f)}$	$2^{(g)}$	1
8. Scram Discharge Volume Water Level - High	1, 2,	2	1
	$5^{(h)}$	2	3
9. Turbine Stop Valve - Closure	$1^{(i)}$	$4^{(j)}$	6
10. Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure - Low	$1^{(i)}$	$2^{(j)}$	6
11. Reactor Mode Switch Shutdown Position	1, 2	1	1
	3, 4	1	7
	5	1	3
12. Manual Scram	1, 2	1	1
	3, 4	1	8
	5	1	9
13. Control Rod Drive			
a. Charging Water Header Pressure - Low	2	2	1
	$5^{(h)}$	2	3
b. Delay Timer	2	2	1
	$5^{(h)}$	2	3

INFO ONLY-NO CHANGES

INFO ONLY - NO CHANGES

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION STATEMENTS

ACTION 1 - Be in at least HOT SHUTDOWN within 12 hours.

ACTION 2 - Verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within 1 hour.

ACTION 3 - Suspend all operations involving CORE ALTERATIONS* and insert all insertable control rods within one hour.

ACTION 4 - Be in at least STARTUP within 6 hours.

ACTION 5 - Be in STARTUP with the main steam line isolation valves closed within 6 hours or in at least HOT SHUTDOWN within 12 hours.

ACTION 6 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure to $\leq$ 140 psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER, within 2 hours.

ACTION 7 - Verify all insertable control rods to be inserted within 1 hour.

ACTION 8 - Lock the reactor mode switch in the Shutdown position within 1 hour.

ACTION 9 - Suspend all operations involving CORE ALTERATIONS,* and insert all insertable control rods and lock the reactor mode switch in the SHUTDOWN position within 1 hour.

---

*Except movement of IRM, SRM, or special movable detectors, or replacement of LPRM strings provided SRM instrumentation is OPERABLE per Specification 3.9.2.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to ~~2~~ 6 hours for required surveillance without placing the channel in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(b) The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn* and during shutdown margin demonstrations performed per Specification 3.10.3.

(c) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than 14 LPRM inputs to an APRM channel.

(d) This function is not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.10.1.

(e) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(f) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(g) Also actuates the standby gas treatment system.

(h) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(i) This function shall be automatically bypassed when turbine first stage pressure is $\leq$ 140 psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER.

(j) Also actuates the EOC-RPT system.

---

*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 3.3.1-2
REACTOR PROTECTION SYSTEM RESPONSE TIMES

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High*	NA
b. Inoperative	NA
2. Average Power Range Monitor*	
a. Neutron Flux - High, Setdown	NA
b. Flow Biased Simulated Thermal Power-Upscale	$\leq$ 0.09**
c. Fixed Neutron Flux - High	$\leq$ 0.09
d. Inoperative	NA
3. Reactor Vessel Steam Dome Pressure - High	$\leq$ 0.55
4. Reactor Vessel Water Level - Low, Level 3	$\leq$ 1.05
5. Main Steam Line Isolation Valve - Closure	$\leq$ 0.06
6. Main Steam Line Radiation - High	NA
7. Primary Containment Pressure - High	NA
8. Scram Discharge Volume Water Level - High	NA
9. Turbine Stop Valve - Closure	$\leq$ 0.06
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	$\leq$ 0.08#
11. Reactor Mode Switch Shutdown Position	NA
12. Manual Scram	NA
13. Control Rod Drive	
a. Charging Water Header Pressure - Low	NA
b. Delay Timer	NA

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel.
**Not including simulated thermal power time constant.
#Measured from start of turbine control valve fast closure.

INFO ONLY - NO CHANGES

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors				
a. Neutron Flux - High	S/U[b], S	S/U[c], W	R	2 *
	S	W	R	3 *, 4, 5
b. Inoperative	NA	W	NA	2 *, 3 *, 4, 5
2. Average Power Range Monitor:[f]				
a. Neutron Flux - High, Setdown	S/U[b], S	S/U[c], W	SA	1, 2 *
	S	W	SA	3 *, 5
b. Flow Biased Simulated Thermal Power-Upscale	S, D[g]	S/U[c], ~~W~~ Q	W[d][e], SA, R[h]	1
c. Fixed Neutron Flux - High	S	S/U[c], ~~W~~ Q	W[d], SA	1
d. Inoperative	NA	~~W~~ Q	NA	1, 2, 3, 5
3. Reactor Vessel Steam Dome Pressure - High	NA	~~W~~ Q	Q	1, 2
4. Reactor Vessel Water Level - Low, Level 3	~~NA~~ S	~~W~~ Q	R	1, 2
5. Main Steam Line Isolation Valve - Closure	NA	Q	R	1
6. Main Steam Line Radiation - High	S	~~W~~ Q	R	1, 2
7. Primary Containment Pressure - High	NA	~~W~~ Q	Q	1, 2

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High	NA	~~M~~ Q	R	1, 2, 5
9. Turbine Stop Valve - Closure	NA	Q	R	1
10. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	NA	Q	R	1
11. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	W	NA	1, 2, 3, 4, 5
13. Control Rod Drive				
a. Charging Water Header Pressure - Low	NA	M	R	2, 5
b. Delay Timer	NA	M	R	2, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decades during each startup and the IRM and APRM channels shall be determined to overlap for at least 1/2 decades during each controlled shutdown, if not performed within the previous 7 days.
(c) Within 24 hours prior to startup, if not performed within the previous 7 days.
(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power levels calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥ 25% of RATED THERMAL POWER. The APRM Gain Adjustment Factor (GAF) for any channel shall be equal to the power value determined by the heat balance divided by the APRM reading for that channel.

Within 2 hours, adjust any APRM channel with a GAF > 1.02. In addition, adjust any APRM channel within 12 hours, (1) if power is greater than or equal to 90% of RATED THERMAL POWER and the APRM channel GAF is < 0.98, or (2) if power is less than 90% of RATED THERMAL POWER and the APRM reading exceeds the power value determined by the heat balance by more than 10% of RATED THERMAL POWER. Until any required APRM adjustment has been accomplished, notification shall be posted on the reactor control panel.

(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.
(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH).
(g) Measure and compare core flow to rated core flow.
(h) This calibration shall consist of verifying the 6 ± 1 second simulated thermal power time constant.

INSERT "C"

ATTACHMENT B

PROPOSED CHANGES TO THE TECHNICAL

SPECIFICATIONS FOR OPERATING LICENSES NPF-11 AND NPF-18

INSERT C

* The provisions of Specification 4.0.4 are not applicable for a period of 24 hours after entering OPERATIONAL CONDITION 2 or 3 when shutting down from OPERATIONAL CONDITION 1.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The isolation actuation instrumentation channels shown in Table 3.3.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.2-2 and with ISOLATION SYSTEM RESPONSE TIME as shown in Table 3.3.2-3.

APPLICABILITY: As shown in Table 3.3.2-1.

ACTION:

a. With an isolation actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel(s) and/or trip system in the tripped condition* within one hour.

c. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition*** within one hour and take the ACTION required by Table 3.3.2-1.

INSERT "D"

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*An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.2-1 for that Trip Function shall be taken. 6

**If more channels are inoperable in one trip system than in the other, select that trip system to place in the tripped condition except when this would cause the Trip Function to occur.

***An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 1 hour or the ACTION required by Table 3.3.2-1 for that Trip Function shall be taken.

ATTACHMENT B

PROPOSED CHANGES TO THE TECHNICAL

SPECIFICATIONS FOR OPERATING LICENSES NPF-11 AND NPF-18

INSERT D

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System Requirement for one trip system, either

   1. Place the inoperable channel(s) and/or trip system in the tripped condition* within

   a) 1 hour for trip functions without an OPERABLE channel,
   b) 12 hours for trip functions common to RPS Instrumentation, and
   c) 24 hours for trip functions not common to RPS Instrumentation.

   or

   2. Take the ACTION required by Table 3.3.2-1.

c. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems,

   1. Place at least one trip system** in the tripped condition*** within one hour, and

   2. a) Place the inoperable channel(s) in the remaining trip system in the tripped condition*** within

      1) 1 hour for trip functions without an OPERABLE channel,
      2) 12 hours for trip functions common to RPS Instrumentation, and
      3) 24 hours for trip functions not common to RPS Instrumentation

      or

      b) Take the ACTION required by Table 3.3.2-1.

TABLE 3.3.2-1 (Continued)

within

ISOLATION ACTUATION INSTRUMENTATION

ACTION STATEMENTS

ACTION 20 - Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN ~~with~~ the next 24 hours.

ACTION 21 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 22 - Close the affected system isolation valves within 1 hour and declare the affected system inoperable.

ACTION 23 - Be in at least STARTUP within 6 hours.

ACTION 24 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within 1 hour.

ACTION 25 - Lock the affected system isolation valves closed within 1 hour and declare the affected system inoperable.

ACTION 26 - Provided that the manual initiation function is OPERABLE for each other group valve, inboard or outboard, as applicable, in each line, restore the manual initiation function to OPERABLE status within 24 hours; otherwise, restore the manual initiation function to OPERABLE status within 8 hours; otherwise:

a. Be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, or

b. Close the affected system isolation valves within the next hour and declare the affected system in operable.

TABLE NOTATIONS

* May be bypassed with reactor steam pressure < 1043 psig and all turbine stop valves closed.

** When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

\# During CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

(a) See Specification 3.6.3, Table 3.6.3-1 for valves in each valve group.

(b) A channel may be placed in an inoperable status for up to ~~2~~ 6 hours for required surveillance without placing the channel in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter. In addition for those trip systems with a design providing only one channel per trip system, the channel may be placed in an inoperable status for up to 8 hours for required surveillance testing without placing the channel in the tripped condition provided that the redundant isolation valve, inboard or outboard, as applicable, in each line is operable and all required actuation instrumentation for that redundant valve is OPERABLE, or place the trip system in the tripped condition.

(c) Also actuates the standby gas treatment system.

(d) A channel is OPERABLE if 2 of 4 instruments in that channel are OPERABLE.

(e) Also actuates secondary containment ventilation isolation dampers per Table 3.6.5.2-1.

(f) Closes only RWCU system inlet outboard valve.

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
A. AUTOMATIC INITIATION				
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1) Low, Level 3	~~NA~~ S	~~M~~ Q	R	1, 2, 3
2) Low Low, Level 2	NA	~~M~~ Q	R	1, 2, 3
3) Low Low Low, Level 1	~~NA~~ S	~~M~~ Q	R	1, 2, 3
b. Drywell Pressure - High	NA	~~M~~ Q	Q	1, 2, 3
c. Main Steam Line				
1) Radiation - High	S	~~M~~ Q	R	1, 2, 3
2) Pressure - Low	NA	~~M~~ Q	Q	1
3) Flow - High	NA	~~M~~ Q	R	1, 2, 3
d. Main Steam Line Tunnel Temperature - High	NA	~~M~~ Q	R	1, 2, 3
e. Condenser Vacuum - Low	NA	~~M~~ Q	Q	1, 2*, 3*
f. Main Steam Line Tunnel Δ Temperature - High	NA	~~M~~ Q	R	1, 2, 3
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Vent Exhaust Plenum Radiation - High	S	~~M~~ Q	R	1, 2, 3 and **
b. Drywell Pressure - High	NA	~~M~~ Q	Q	1, 2, 3
c. Reactor Vessel Water Level - Low Low, Level 2	NA	~~M~~ Q	R	1, 2, 3, and #
d. Fuel Pool Vent Exhaust Radiation - High	S	~~M~~ Q	R	1, 2, 3 and **
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	S	~~M~~ Q	R	1, 2, 3
b. Heat Exchanger Area Temperature - High	NA	~~M~~ Q	Q	1, 2, 3
c. Heat Exchanger Area Ventilation ΔT - High	NA	~~M~~ Q	Q	1, 2, 3
d. SLCS Initiation	NA	R	NA	1, 2, 3
e. Reactor Vessel Water Level - Low Low, Level 2	NA	~~M~~ Q	R	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
4. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow - High	NA	~~M~~ Q	Q	1, 2, 3
b. RCIC Steam Supply Pressure - Low	NA	~~M~~ Q	Q	1, 2, 3
c. RCIC Turbine Exhaust Diaphragm Pressure - High	NA	~~M~~ Q	Q	1, 2, 3
d. RCIC Equipment Room Temperature - High	NA	~~M~~ Q	Q	1, 2, 3
e. RCIC Steam Line Tunnel Temperature - High	NA	~~M~~ Q	Q	1, 2, 3
f. RCIC Steam Line Tunnel Δ W Temperature - High	NA	~~M~~ Q	Q	1, 2, 3
g. Drywell Pressure - High	NA	~~M~~ Q	Q	1, 2, 3
h. RCIC Equipment Room Δ Temperature - High	NA	~~M~~ Q	Q	1, 2, 3
5. RHR SYSTEM STEAM CONDENSING MODE ISOLATION				
a. RHR Equipment Area W Δ Temperature - High	NA	~~M~~ Q	Q	1, 2, 3
b. RHR Area Cooler Temperature - High	NA	~~M~~ Q	Q	1, 2, 3
c. RHR Heat Exchanger Steam Supply Flow - High	NA	~~M~~ Q	Q	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3	~~NA~~ S	~~M~~ Q	R	1, 2, 3
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA	~~M~~ Q	Q	1, 2, 3
c. RHR Pump Suction Flow - High	NA	~~M~~ Q	Q	1, 2, 3
d. RHR Area Temperature - High	NA	~~M~~ Q	Q	1, 2, 3
e. RHR Equipment Area ΔT - High	NA	~~M~~ Q	Q	1, 2, 3
B. MANUAL INITIATION				
1. Inboard Valves	NA	R	NA	1, 2, 3
2. Outboard Valves	NA	R	NA	1, 2, 3
3. Inboard Valves	NA	R	NA	1, 2, 3 and **,#
4. Outboard Valves	NA	R	NA	1, 2, 3 and **,#
5. Inboard Valves	NA	R	NA	1, 2, 3
6. Outboard Valves	NA	R	NA	1, 2, 3
7. Outboard Valve	NA	R	NA	1, 2, 3

*When reactor steam pressure > 1043 psig and/or any turbine stop valve is open.

**When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

#During CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION(a)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
C. DIVISION 3 TRIP SYSTEM			
1. HPCS SYSTEM			
a. Reactor Vessel Water Level - Low, Low, Level 2	4(b)	1, 2, 3, 4*, 5*	35
b. Drywell Pressure - High	4(b)	1, 2, 3	35
c. Reactor Vessel Water Level-High, Level 8	2(c)	1, 2, 3, 4*, 5*	32
d. Deleted			
e. Deleted			
f. Pump Discharge Pressure-High (Bypass)	1	1, 2, 3, 4*, 5*	31
g. HPCS System Flow Rate-Low (Permissive)	1	1, 2, 3, 4*, 5*	31
h. Manual Initiation	1/division	1, 2, 3, 4*, 5*	34

D. LOSS OF POWER	TOTAL NO. OF INSTRUMENTS	INSTRUMENTS TO TRIP	MINIMUM OPERABLE INSTRUMENTS~~(a)~~ (d)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	2/bus	2/bus	2/bus	1, 2, 3, 4**, 5**	37
2. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	2/bus	2/bus	2/bus	1, 2, 3, 4**, 5**	37

TABLE NOTATION

(a) A channel/instrument may be placed in an inoperable status for up to ~~2~~ 6 hours during periods of required surveillance without placing the trip system/channel/instrument in the tripped condition provided at least one other OPERABLE channel/instrument in the same trip system is monitoring that parameter.
(b) Also actuates the associated division diesel generator.
(c) Provides signal to close HPCS pump discharge valve only on 2-out-of-2 logic.
* Applicable when the system is required to be OPERABLE per Specification 3.5.2 or 3.5.3.
** Required when ESF equipment is required to be OPERABLE.
\# Not required to be OPERABLE when reactor steam dome pressure is ≤ 122 psig.

INSERT "E"

ATTACHMENT B

PROPOSED CHANGES TO THE TECHNICAL

SPECIFICATIONS FOR OPERATING LICENSES NPF-11 AND NPF-18

INSERT E

(d) A channel/instrument may be placed in an inoperable status for up to 2 hours during periods of required surveillance without placing the trip system/channel/instrument in the tripped condition provided at least one other OPERABLE channel/instrument in the same trip system is monitoring that parameter.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION

ACTION 30 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

a. With one channel inoperable, place the inoperable channel in the tripped condition within one hour or declare the associated system inoperable. 24 hours

b. With more than one channel inoperable, declare the associated system inoperable. requirement

ACTION 31 - With the number of OPERABLE channels less than required by the Minimum OPERABLE channels per Trip Function, place the inoperable channel in the tripped condition within one hour; restore the inoperable channel to OPERABLE status within 7 days or declare the associated system inoperable. 24 hours

ACTION 32 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, declare the associated ADS trip system or ECCS inoperable. within 24 hours

ACTION 33 - With the number of OPERABLE channels less than the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour. 24 hours

ACTION 34 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 8 hours or declare the associated ADS trip system or ECCS inoperable. 24

ACTION 35 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement

a. For one trip system, place that trip system in the tripped condition within one hour or declare the HPCS system inoperable. 24 hours

b. For both trip systems, declare the HPCS system inoperable.

ACTION 36 - Deleted

ACTION 37 - With the number of OPERABLE instruments less than the Minimum Operable Instruments, place the inoperable instrument(s) in the tripped condition within 1 hour or declare the associated emergency diesel generator inoperable and take the ACTION required by Specification 3.8.1.1 or 3.8.1.2 as appropriate.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION

ACTION 38 With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per trip function requirements:

a. With one channel inoperable, remove the inoperable channel within one hour; restore the inoperable channel to OPERABLE status within 7 days or declare the associated ECCS systems inoperable.

24 hours

b. With both channels inoperable, restore at least one channel to OPERABLE status within one hour or declare the associated ECCS system inoperable.

TABLE 4.3.3.1-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
A. DIVISION I TRIP SYSTEM				
1. RHR-A (LPCI MODE) AND LPCS SYSTEM				
a. Reactor Vessel Water Level - Low Low Low, Level 1	~~NA~~ S	~~M~~ Q	R	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	~~NA~~	~~M~~ Q	Q	1, 2, 3
c. LPCS Pump Discharge Flow-Low	NA	~~M~~ Q	Q	1, 2, 3, 4*, 5*
d. LPCS and LPCI A Injection Valve Injection Line Pressure Low Interlock	NA	~~M~~ Q	R	1, 2, 3, 4*, 5*
e. LPCS and LCPI A Injection Valve Reactor Pressure Low Interlock	NA	~~M~~ Q	R	1, 2, 3, 4*, 5*
f. LPCI Pump A Start Time Delay Relay	NA	~~M~~ Q	Q	1, 2, 3, 4*, 5*
g. LPCI Pump A Flow-Low	NA	~~M~~ Q	Q	1, 2, 3, 4*, 5*
h. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"#				
a. Reactor Vessel Water Level - Low Low Low, Level 1	~~NA~~ S	~~M~~ Q	R	1, 2, 3
b. Drywell Pressure-High	~~NA~~	~~M~~ Q	Q	1, 2, 3
c. Initiation Timer	NA	~~M~~ Q	Q	1, 2, 3
d. Reactor Vessel Water Level - Low, Level 3	~~NA~~ S	~~M~~ Q	R	1, 2, 3
e. LPCS Pump Discharge Pressure-High	NA	~~M~~ Q	Q	1, 2, 3
f. LPCI Pump A Discharge Pressure-High	NA	~~M~~ Q	Q	1, 2, 3
g. Manual Initiation	NA	R	NA	1, 2, 3
h. Drywell Pressure Bypass Timer	NA	~~M~~ Q	Q	1, 2, 3
i. Manual Inhibit	NA	R	NA	1, 2, 3

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
B. DIVISION 2 TRIP SYSTEM				
1. RHR B AND C (LPCI MODE)				
a. Reactor Vessel Water Level - Low Low Low, Level 1	~~NA~~ S	~~M~~ Q	R	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	NA	~~M~~ Q	Q	1, 2, 3
c. LPCI B and C Injection Valve Injection Line Pressure Low Interlock	NA	~~M~~ Q	R	1, 2, 3, 4*, 5*
d. LPCI Pump B Start Time Delay Relay	NA	~~M~~ Q	Q	1, 2, 3, 4*, 5*
e. LPCI Pump Discharge Flow-Low	NA	~~M~~ Q	Q	1, 2, 3, 4*, 5*
f. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
g. LPCI B and C Injection Valve Reactor Pressure Low Interlock	NA	~~M~~ Q	R	1, 2, 3, 4*, 5*
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"#				
a. Reactor Vessel Water Level - Low Low Low, Level 1	~~NA~~ S	~~M~~ Q	R	1, 2, 3
b. Drywell Pressure-High	NA	~~M~~ Q	Q	1, 2, 3
c. Initiation Timer	NA	~~M~~ Q	Q	1, 2, 3
d. Reactor Vessel Water Level - Low, Level 3	~~NA~~ S	~~M~~ Q	R	1, 2, 3
e. LPCI Pump B and C Discharge Pressure-High	NA	~~M~~ Q	Q	1, 2, 3
f. Manual Initiation	NA	R	NA	1, 2, 3
g. Drywell Pressure Bypass Timer	NA	~~M~~ Q	Q	1, 2, 3
h. Manual Inhibit	NA	R	NA	1, 2, 3

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
C. DIVISION 3 TRIP SYSTEM				
1. HPCS SYSTEM				
a. Reactor Vessel Water Level - Low Low, Level 2	~~NA~~ S	~~M~~ Q	R	1, 2, 3, 4*, 5*
b. Drywell Pressure-High	NA	~~M~~ Q	Q	1, 2, 3
c. Reactor Vessel Water Level-High Level 8	~~NA~~ S	~~M~~ Q	R	1, 2, 3, 4*, 5*
d. Deleted				
e. Deleted				
f. Pump Discharge Pressure-High	NA	~~M~~ Q	Q	1, 2, 3, 4*, 5*
g. HPCS System Flow Rate-Low	NA	~~M~~ Q	Q	1, 2, 3, 4*, 5*
h. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
D. LOSS OF POWER				
1. 4.16 kV Emergency Bus Under-voltage (Loss of Voltage)	NA	NA	R	1, 2, 3, 4**, 5**
2. 4.16 kV Emergency Bus Under-voltage (Degraded Voltage)	NA	NA	R	1, 2, 3, 4**, 5**

TABLE NOTATIONS

#Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 122 psig.

*When the system is required to be OPERABLE after being manually realigned, as applicable, per Specification 3.5.2.

**Required when ESF equipment is required to be OPERABLE.

INSTRUMENTATION

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.1 The anticipated transient without scram recirculation pump trip (ATWS-RPT) system instrumentation channels shown in Table 3.3.4.1-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4.1-2.

APPLICABILITY: OPERATIONAL CONDITION 1.

ACTION:

a. With an ATWS recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.1-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within 1 hour. 24 hours

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system and: 24 hours

   1. If the inoperable channels consist of one reactor vessel water level channel and one reactor vessel pressure channel, place both inoperable channels in the tripped condition within 1 hour, or, if this action will initiate a pump trip, declare the trip system inoperable.

   2. If the inoperable channels include two reactor vessel water level channels or two reactor vessel pressure channels, declare the trip system inoperable.

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within 1 hour or be in at least STARTUP within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.3.4.1.1 Each ATWS recirculation pump trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.4.1-1.

4.3.4.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

TABLE 3.3.4.1-1

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)
1. Reactor Vessel Water Level - Low Low, Level 2	2
2. Reactor Vessel Pressure-High	2

(a) One channel in one trip system may be placed in an inoperable status for up to ~~2~~ 6 hours for required surveillance provided that all other channels are OPERABLE.

TABLE 3.3.4.1-2

ATWS RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel, Water Level - Low Low, Level 2	≥ -50 inches*	≥ -57 inches*
2. Reactor Vessel Pressure-High	≤ 1135 psig	≤ 1150 psig

* See Bases Figure B3/4 3-1.

INFO ONLY- NO CHANGES

TABLE 4.3.4.1-1

ATWS RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Reactor Vessel Water Level - Low Low, Level 2	S	~~M~~ Q	R
2. Reactor Vessel Pressure - High	S	~~M~~ Q	Q

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.2 The end-of-cycle recirculation pump trip (EOC-RPT) system instrumentation channels shown in Table 3.3.4.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4.2-2 and with the END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME as shown in Table 3.3.4.2-3.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER.

ACTION:

a. With an end-of-cycle recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within 1 hour. 12 hours

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement(s) for one trip system and:

   1. If the inoperable channels consist of one turbine control valve channel and one turbine stop valve channel, place both inoperable channels in the tripped condition within 1 hour. 12 hours

   2. If the inoperable channels include two turbine control valve channels or two turbine stop valve channels, declare the trip system inoperable.

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours, otherwise, either:

   1. Increase the MINIMUM CRITICAL POWER (MCPR) Limiting Condition for Operation (LCO) to the EOC-RPT inoperable value per Specification 3.2.3 within the next 1 hour, or

   2. Reduce THERMAL POWER to less than 30% of RATED THERMAL POWER within the next 6 hours.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within 1 hour,

   1. Increase the MINIMUM CRITICAL POWER (MCPR) Limiting Condition for Operation (LCO) to the EOC-RPT inoperable value per Specification 3.2.3 within the next 1 hour, or

   2. Reduce THERMAL POWER to less than 30% RATED THERMAL POWER within the next 6 hours.

INFO ONLY - NO CHANGES

SURVEILLANCE REQUIREMENTS

4.3.4.2.1 Each end-of-cycle recirculation pump trip system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.4.2.1-1.

4.3.4.2.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.4.2.3 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME of each trip function shown in Table 3.3.4.2-3 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least the logic of one type of channel input, turbine control valve fast closure or turbine stop valve closure, such that both types of channel inputs are tested at least once per 36 months. The time allotted for breaker arc suppression shall be verified by test at least once per 60 months.

TABLE 3.3.4.2-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[(a)]
1. Turbine Stop Valve - Closure	2[(b)]
2. Turbine Control Valve - Fast Closure	2[(b)]

(a) A trip system may be placed in an inoperable status for up to 2 6 hours for required surveillance provided that the other trip system is OPERABLE.

(b) This function shall be automatically bypassed when turbine first stage pressure is less than or equal to 140 psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER.

TABLE 4.3.4.2.1-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Turbine Stop Valve Closure	Q	R
2. Turbine Control Valve-Fast Closure	Q	R

INFO ONLY - NO CHANGES

INFO ONLY - NO CHANGES

INSTRUMENTATION

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.5 The reactor core isolation cooling (RCIC) system actuation instrumentation channels shown in Table 3.3.5-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.5-2.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3 with reactor steam dome pressure greater than 150 psig.

ACTION:

a. With a RCIC system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.5-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more RCIC system actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.5-1.

SURVEILLANCE REQUIREMENTS

4.3.5.1 Each RCIC system actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.5.1-1.

4.3.5.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

TABLE 3.3.5-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

FUNCTIONAL UNITS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM(a)	ACTION
a. Reactor Vessel Water Level - Low Low, Level 2	2	50
b. Reactor Vessel Water Level - High, Level 8	2(b)	51
c. Manual Initiation	1(c)	52

(a) A channel may be placed in an inoperable status for up to ~~2~~ 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

(b) One trip system with two-out-of-two logic.

(c) Single channel.

TABLE 3.3.5-1 (Continued)

REACTOR CORE ISOLATION COOLING SYSTEM
ACTUATION INSTRUMENTATION

ACTION 50 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement:

a. For one trip system, place the inoperable channel in the tripped condition within one hour or declare the RCIC system inoperable. 24 hours

b. For both trip systems, declare the RCIC system inoperable.

ACTION 51 - With the number of OPERABLE channel less than required by the minimum OPERABLE Channels per Trip System requirement, declare the RCIC system inoperable within 24 hours.

ACTION 52 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, restore the inoperable channel to OPERABLE status within 8 hours or declare the RCIC system inoperable.
24

TABLE 4.3.5.1-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNITS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
a. Reactor Vessel Water Level - Low Low, Level 2	NA	~~M~~ Q	R
b. Reactor Vessel Water Level - High, Level 8	~~NA~~ S	~~M~~ Q	R
c. Manual Initiation	NA	R	NA

INSTRUMENTATION

3/4.3.6 CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6 The control rod withdrawal block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

a. With a control rod withdrawal block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, take the ACTION required by Table 3.3.6-1.

SURVEILLANCE REQUIREMENTS

4.3.6 Each of the above required control rod withdrawal block trip systems and instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.6-1. *

INSERT "F"

ATTACHMENT B

PROPOSED CHANGES TO THE TECHNICAL

SPECIFICATIONS FOR OPERATING LICENSES NPF-11 AND NPF-18

INSERT F

* A channel may be placed in an inoperable status for up to 6 hours for required surveillance (or 12 hours for repair) without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

TABLE 3.3.6-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

ACTION

ACTION 60 - Declare the RBM inoperable and take the ACTION required by Specification 3.1.4.3.

ACTION 61 - With the number of OPERABLE channels:

a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 7 days or place the inoperable channel in the tripped condition within the next hour.

b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.

ACTION 62 - With the number of OPERABLE Channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within ~~one hour~~. 12 hours

NOTE

* With THERMAL POWER ≥ 30% of RATED THERMAL POWER.

** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

a. The RBM shall be automatically bypassed when a peripheral control rod is selected.

b. This function shall be automatically bypassed if detector count rate is ≥ 100 cps or the IRM channels are on range 3 or higher.

c. This function shall be automatically bypassed when the associated IRM channels are on range 8 or higher.

d. This function shall be automatically bypassed when the IRM channels are on range 3 or higher.

e. This function shall be automatically bypassed when the IRM channels are on range 1.

TABLE 4.3.6-1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	N.A.	S/U(b)(c), ~~M~~(c) Q	Q	1*
b. Inoperative	N.A.	S/U(b)(c), ~~M~~(c) Q	N.A.	1*
c. Downscale	N.A.	S/U(b)(c), ~~M~~(c) Q	Q	1*
2. APRM				
a. Flow Biased Simulated Thermal Power-Upscale	N.A.	S/U(b), ~~M~~ Q	SA	1
b. Inoperative	N.A.	S/U(b), ~~M~~ Q	N.A.	1, 2, 5
c. Downscale	N.A.	S/U(b), ~~M~~ Q	SA	1
d. Neutron Flux-High	N.A.	S/U(b), ~~M~~ Q	SA	2***, 5
3. SOURCE RANGE MONITORS				
a. Detector not full in	N.A.	S/U(b), W	N.A.	2***, 5
b. Upscale	N.A.	S/U(b), W	Q	2***, 5
c. Inoperative	N.A.	S/U(b), W	N.A.	2***, 5
d. Downscale	N.A.	S/U(b), W	Q	2***, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	N.A.	S/U(b), W	N.A.	2***, 5
b. Upscale	N.A.	S/U(b), W	Q	2***, 5
c. Inoperative	N.A.	S/U(b), W	N.A.	2***, 5
d. Downscale	N.A.	S/U(b), W	Q	2***, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	N.A.	Q	R	1, 2, 5**
b. Scram Discharge Volume Switch in Bypass	N.A.	~~M~~ Q	N.A.	5**
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	N.A.	S/U(b), ~~M~~ Q	Q	1
b. Inoperative	N.A.	S/U(b), ~~M~~ Q	N.A.	1
c. Comparator	N.A.	S/U(b), ~~M~~ Q	Q	1

TABLE 4.3.6-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATIONS

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) Within 24 hours prior to startup, if not performed within the previous 7 days.

(c) Includes reactor manual control multiplexing system input.

*With THERMAL POWER ≥ 30% of RATED THERMAL POWER.

**With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

INSERT "G"

ATTACHMENT B

PROPOSED CHANGES TO THE TECHNICAL

SPECIFICATIONS FOR OPERATING LICENSES NPF-11 AND NPF-18

INSERT G

*** The provisions of Specification 4.0.4 are not applicable for a period of 24 hours after entering OPERATIONAL CONDITION 2 or 3 when shutting down from OPERATIONAL CONDITION 1.

 INFO ONLY - NO CHANGES

3/4.3.7 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.7.1 The radiation monitoring instrumentation channels shown in Table 3.3.7.1-1 shall be OPERABLE* with their alarm/trip setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3.7.1-1.

ACTION:

a. With a radiation monitoring instrumentation channel alarm/trip setpoint exceeding the value shown in Table 3.3.7.1-1, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels inoperable, take the ACTION required by Table 3.3.7.1-1.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.7.1 Each of the above required radiation monitoring instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the conditions and at the frequencies shown in Table 4.3.7.1-1.

*The normal or emergency power source may be inoperable in OPERATIONAL CONDITION 4 or 5 or when defueled.

TABLE 3.3.7.1-1

RADIATION MONITORING INSTRUMENTATION

INSTRUMENTATION	MINIMUM CHANNELS OPERABLE	APPLICABLE CONDITIONS	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
a. Main Control Room Atmospheric Control System Radiation Monitoring Subsystem	2/intake**	1,2,3,5 and *	3.[illegible]/hr	0.1 to 10,000 mR/hr	70

TABLE NOTATIONS

*When irradiated fuel is being handled in the secondary containment.

INSERT "H"

ACTION STATEMENT

ACTION 70 -

a. With one of the required monitors inoperable, place the inoperable channel in the downscale tripped condition within 1 hour; restore the inoperable channel to OPERABLE status within 7 days, or, within the next 6 hours, initiate and maintain operation of the control room emergency filtration system in the pressurization mode of operation.

b. With both of the required monitors inoperable, initiate and maintain operation of the control room emergency filtration system in the pressurization mode of operation within 1 hour.

ATTACHMENT B

PROPOSED CHANGES TO THE TECHNICAL

SPECIFICATIONS FOR OPERATING LICENSES NPF-11 AND NPF-18

INSERT H

** A channel may be placed in an inoperable status for up to 6 hours for required surveillance testing without placing the Trip System in the tripped condition, provided at least one other operable channel in the same Trip System is monitoring that Trip Function.

TABLE 4.3.7.1-1

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTATION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
a. Main Control Room Atmospheric Control System Radiation Monitoring Subsystem	S	Q	R	1,2,3,5 and *

NOTES

*When irradiated fuel is being handled in the secondary containment.

3/4.3.8 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.8 The feedwater/main turbine trip system actuation instrumentation channels shown in Table 3.3.8-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.8-2.

APPLICABILITY: OPERATIONAL CONDITION 1.

INSERT "I"

ACTION:

a. With a feedwater/main turbine trip system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.8-2, declare the channel inoperable and either place the inoperable channel in the tripped condition until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value, or declare the associated system inoperable.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement, restore the inoperable channel to OPERABLE status within 7 days or be in at least STARTUP within the next 6 hours.

c. With the number of OPERABLE channels two less than required by the Minimum OPERABLE Channels per Trip System requirement, restore at least one of the inoperable channels to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.3.8.1 Each feedwater/main turbine trip system actuation instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.3.8.1-1.

4.3.8.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

INSERT "J"

ATTACHMENT B

PROPOSED CHANGES TO THE TECHNICAL

SPECIFICATIONS FOR OPERATING LICENSES NPF-11 AND NPF-18

INSERT I

a. With a feedwater/main turbine trip system actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.8-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than that required by the Minimum OPERABLE Channels per Trip System requirement:

1. Within 7 days, either place the inoperable channel in the tripped* condition or restore the inoperable channel to OPERABLE status.

2. Otherwise, be in at least STARTUP within 6 hours.

c. With the number of OPERABLE channels two less than required by the Minimum OPERABLE Channels per Trip System requirement:

1. Within two hours place or verify at least one inoperable channel in the tripped* condition, and restore either inoperable channel to OPERABLE status within 72 hours, or,

2. Be in at least STARTUP within the next 6 hours.

INSERT J

* An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur.

TABLE 3.3.8-1

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM
a. Reactor Vessel Water Level-High, Level 8	3*

INSERT "K"

ATTACHMENT B

PROPOSED CHANGES TO THE TECHNICAL

SPECIFICATIONS FOR OPERATING LICENSES NPF-11 AND NPF-18

INSERT K

* A channel may be placed in an inoperable status for up to 6 hours for required surveillance testing without placing the Trip System in the tripped condition.

TABLE 3.3.8-2

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
a. Reactor Vessel Water Level-High, Level 8	< 55.5 inches*	< 56.0 inches*

INFO ONLY-NO CHANGES

*See Bases Figure B 3/4 3-1.

TABLE 4.3.8.1-1

FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
a. Reactor Vessel Water Level-High, Level 8	S	Q	R

3/4.3 INSTRUMENTATION

BASES

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

a. Preserve the integrity of the fuel cladding.

b. Preserve the integrity of the reactor coolant system.

c. Minimize the energy which must be adsorbed following a loss-of-coolant accident, and

d. Prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. The system meets the intent of IEEE-279, 1971, for nuclear power plant protection systems. Specified surveillance intervals for MSIV-Closure, TSV-Closure, TCV-Closure, and the Manual Scram have been determined in accordance with NEDC-30851P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System", March 1988. The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1. INSERT "M"

The measurement of response time at the specified frequencies provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the accident analysis. No credit was taken for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurement, provided such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) inplace, onsite or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

and surveillance and maintenance outage times

INSERT "L"

ATTACHMENT B

PROPOSED CHANGES TO THE TECHNICAL

SPECIFICATIONS FOR OPERATING LICENSES NPF-11 AND NPF-18

INSERT L

and MDE-83-0485 Revision 3, "Technical Specification Improvement Analysis for the Reactor Protection System for LaSalle County Station, Units 1 and 2", April 1991.

INSERT M

When a channel is placed in an inoperable status solely for performance of required surveillances, entry into LCO and required ACTIONS may be delayed, provided the associated function maintains RPS trip capability.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

INSERT "N"

This specification ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents by prescribing the OPERABILITY trip setpoints and response times for isolation of the reactor systems. When necessary, one channel may be inoperable for brief intervals to conduct required surveillance. Both channels of each trip system for the main steam tunnel ambient temperature and ventilation system differential temperature may be placed in an inoperable status for up to 4 hours for required reactor building ventilation system maintenance and testing and 12 hours for the required secondary containment Leak Rate test without placing the trip system in the tripped condition. This will allow for maintaining the reliability of the ventilation system and secondary containment. Some of the trip settings may have tolerances explicitly stated where both the high and low values are critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting have a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

Except for the MSIVs, the safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. For A.C. operated valves, it is assumed that the A.C. power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of 13 seconds is assumed before the valve starts to move. The safety analysis considers an allowable inventory loss which in turn determines the valve speed in conjunction with the 13 second delay.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

The emergency core cooling system actuation instrumentation is provided to initiate actions to mitigate the consequences of accidents that are beyond the ability of the operator to control. This specification provides the OPERABILITY requirements, trip setpoints and response times that will ensure effectiveness of the systems to provide the design protection. Although the instruments are listed by system, in some cases the same instrument may be used to send the actuation signal to more than one system at the same time.

INSERT "O"

ATTACHMENT B

PROPOSED CHANGES TO THE TECHNICAL

SPECIFICATIONS FOR OPERATING LICENSES NPF-11 AND NPF-18

INSERT N

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P-A, Supplement 2, "Technical Specification Improvement Analyses for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation", March 1989, and with NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation", July 1990. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into LCO and required ACTIONS may be delayed, provided the associated function maintains primary containment isolation capability.

INSERT O

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30936P-A, "Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation)", Parts 1 and 2, December 1988, and RE-025 Revision 1, "Technical Specification Improvement Analysis for the Emergency Core Cooling System Actuation Instrumentation for LaSalle County Station, Units 1 and 2", April 1991. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into LCO and required ACTIONS may be delayed, provided the associated function maintains ECCS initiation capability.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971 and NEDO-24222, dated December, 1979, and Appendix G of the FSAR.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a part of the Reactor Protection System and is an essential safety supplement to the reactor trip. The purpose of the EOC-RPT is to recover the loss of thermal margin which occurs at the end-of-cycle. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity to the reactor system at a faster rate than the control rods add negative scram reactivity. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

A generic analysis, which provides for continued operation with one or both trip systems of the EOC-RPT system inoperable, has been performed. The analysis determined bounding cycle independent MINIMUM CRITICAL POWER RATIO (MCPR) Limiting Condition for Operation (LCO) values which must be used if the EOC-RPT system is inoperable. These values ensure that adequate reactivity margin to the MCPR safety limit exists in the event of the analyzed transient with the RPT function inoperable. The analysis results are further discussed in the bases for Specification 3.2.3.

A fast closure sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a position switch for each of two turbine stop valves provides input to one EOC-RPT system; a position switch from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 30% of RATED THERMAL POWER are annunciated in the control room.

and surveillance and maintenance outage times

Specified surveillance intervals have been determined in accordance with the following:

1. NEDC-30851P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System", March 1988.

INFO ONLY - NO CHANGES

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

2. GENE-770-06-1-A, "Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications", December 1992.

The EOC-RPT system response time is the time assumed in the analysis between initiation of valve motion and complete suppression of the electric arc, i.e., 190 ms, less the time allotted for sensor response, i.e., 10 ms, and less the time allotted for breaker arc suppression determined by test, as correlated to manufacturer's test results, i.e., 83 ms, and plant pre-operational test results.

INSERT "P"

ATTACHMENT B

PROPOSED CHANGES TO THE TECHNICAL

SPECIFICATIONS FOR OPERATING LICENSES NPF-11 AND NPF-18

INSERT P

When a channel is placed in an inoperable status solely for performance of required surveillances, entry into LCO and required ACTIONS may be delayed provided the associated function maintains the applicable RPT initiation capability.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without providing actuation of any of the emergency core cooling equipment.

INSERT "Q"

3/4.3.6 CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of the specifications in Section 3/4.1.4, Control Rod Program Controls and Section 3/4.2 Power Distribution Limits. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

INSERT "R"

3/4.3.7 MONITORING INSTRUMENTATION

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring instrumentation ensures that; (1) the radiation levels are continually measured in the areas served by the individual channels, and (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

INSERT "S"

3.4.3.7.2 SEISMIC MONITORING INSTRUMENTATION

The OPERABILITY of the seismic monitoring instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the unit. This instrumentation is consistent with the recommendations of Regulatory Guide 1.12 "Instrumentation for Earthquakes", April 1974.

3/4.3.7.3 METEOROLOGICAL MONITORING INSTRUMENTATION

The OPERABILITY of the meteorological monitoring instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public. This instrumentation is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," February, 1972.

3/4.3.7.4 REMOTE SHUTDOWN MONITORING INSTRUMENTATION

The OPERABILITY of the remote shutdown monitoring instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN of the unit from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

ATTACHMENT B

PROPOSED CHANGES TO THE TECHNICAL

SPECIFICATIONS FOR OPERATING LICENSES NPF-11 AND NPF-18

INSERT Q

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with GENE-770-06-2-A, "Addendum To Bases for Changes to Surveillance Test Intervals and Allowed Out-of-Service Times for Selected Instrumentation Technical Specifications (BWR RCIC Instrumentation)", December 1992. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into LCO and required ACTIONS may be delayed, provided the associated function maintains RCIC initiation capability.

INSERT R

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P-A, Supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation", October 1988, and GENE-770-06-1-A, "Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times for Selected Instrumentation Technical Specifications", December 1992. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into LCO and required ACTIONS may be delayed, provided the associated function maintains Control Rod Block capability.

INSERT S

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with GENE-770-06-1-A, "Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times for Selected Instrumentation Technical Specifications", December 1992. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into LCO and required ACTIONS may be delayed, provided the associated function maintains initiation capability.

3/4.3.7.11 EXPLOSIVE GAS MONITORING INSTRUMENTATION

This instrumentation provides for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the waste gas holdup system.

3/4.3.7.12 LOOSE-PART DETECTION SYSTEM

The OPERABILITY of the loose-part detection system ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors."

3/4.3.8 FEEDWATER/MAIN TURBINE TRIP SYSTEM ACTUATION INSTRUMENTATION

The feedwater/main turbine trip system actuation instrumentation is provided to initiate the feedwater system/main turbine trip system in the event of reactor vessel water level equal to or greater than the level 8 setpoint associated with a feedwater controller failure to prevent overfilling the reactor vessel which may result in high pressure liquid discharge through the safety/relief valve discharge lines.

INSERT "T"

ATTACHMENT B

PROPOSED CHANGES TO THE TECHNICAL

SPECIFICATIONS FOR OPERATING LICENSES NPF-11 AND NPF-18

INSERT T

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with GENE-770-06-1-A, "Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times for Selected Instrumentation Technical Specifications", December 1992. When a channel is placed in an inoperable status solely for performance of required surveillances, entry into LCO and required ACTIONS may be delayed, provided the associated function maintains Feedwater System/Main Turbine Trip System actuation capability.

ATTACHMENT C
SIGNIFICANT HAZARDS CONSIDERATION

ComEd has evaluated the proposed Technical Specification Amendment for LaSalle County Station Units 1 and 2, which: a) Extends the STI and AOT for certain actuation instrumentation in the reactor protection, isolation, emergency core cooling, recirculation pump trip, reactor core isolation cooling, control rod withdrawal block, monitoring, and feedwater/main turbine trip systems; b) Proposes a change to the Feedwater/Main Turbine Trip LCO action statements to achieve consistency with presently existing instrumentation LCOs; c) Deletes the surveillance of the APRM Neutron Flux - High, Setdown functional unit in Operational Condition 1; d) Revising the applicability of the Provisions of Specification 4.0.4 to several Reactor Protection System and Control Rod Withdrawal Block Instrumentation Surveillance Requirements; e) Add the requirement for performing a shiftly channel check for the applicable RPS, PCIS, ECCS, and RCIC instrumentation channels equipped with master trip units; and, f) Administrative changes to correct typographical errors and to delete cycle specific footnotes which are no longer applicable. It has been determined that the changes do not constitute a Significant Hazards Consideration. Based on the criteria for defining a significant hazards consideration established in 10CFR50.9 , operation of LaSalle County Station Units 1 and 2 in accordance wit the proposed amendment will not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated because:

   a. The proposed changes increase the STI and AOT for actuation instrumentation supporting RPS, ECCS, Isolation, CRBF, RCIC, ATWS-RPT, EOC-RPT, Monitoring, and Feedwater/Main Turbine Trip System Actuation functions.

      There are no changes in instrumentation configuration and function, and no instrumentation setpoints are changed. Because of this there is no change in the probability of occurrence of an accident or the consequences of an accident or the consequences of malfunction of equipment. With respect to the probability of equipment malfunction, topical reports prepared by GE demonstrate that there is a reduction in scram frequency for the RPS, but in the case of the ECCS there is a small increase in the unavailability of the water injection function. This increase in unavailability was judged acceptable by GE. The NRC concurred with this conclusion in its review and approval of the topical reports. The proposed changes are consistent with the Safety Evaluation Reports issued for the topical reports.

ATTACHMENT C
SIGNIFICANT HAZARDS CONSIDERATION

b. The changes proposed for the Feedwater/Main Turbine Trip LCO action statements provide actions which are consistent with presently existing instrumentation LCOs. The design and function of the feedwater/main turbine trip instrumentation to trip the feedwater pumps and the main turbine upon detection of a Level 8 event is not altered. The probability and/or consequences of this moderate frequency transient are not increased.

c. The APRM Neutron Flux - High, Setdown scram setting provides adequate thermal margin between the setpoint and the safety limits for operation at low pressure and low flow during a plant startup. This function remains in effect until the mode switch is placed in the Run (Operational Condition 1) position, at which time it is bypassed. Deleting the requirement for the surveillance of the APRM Neutron Flux - High, Setdown functional unit in Operational Condition 1 is appropriate since its function is not applicable in this mode. This deletion serves to achieve consistency between Technical Specification Tables and the Bases section.

d. The changes associated with Specification 4.0.4 are administrative in nature and are intended to provide the plant operators with better guidance for its application. In cases where complete surveillances cannot be achieved, such as during a plant shutdown, then the required surveillances will be performed within 24 hours of entering the Mode or condition in which the surveillance is required. The stabilization of the plant will be of primary consideration. This change does not affect the evaluation for any accident presented in Chapter 15 of the UFSAR. The APRM Fixed Neutron Flux - High quarterly functional tests most of the APRM channel equipment associated with the APRM Neutron Flux - High, Setdown scram.

Additionally, the expected result of the functional tests associated with the SRMs, IRMs, and APRMs is to demonstrate the operability of the instrumentation. Therefore, 24 hours is a reasonable time to permit the surveillances to be performed upon entering the mode or condition in which the surveillance is required.

e. The proposal to include the performance of channel checks as requirements of technical specifications is administrative in nature. Presently, channel checks performed for the applicable analog instrumentation in reactor vessel water level applications is controlled solely by procedure. Adding this requirement to the technical specifications provides for the appropriate controls of the surveillances, above and beyond that presently controlled by procedure.

ATTACHMENT C
SIGNIFICANT HAZARDS CONSIDERATION

f. The proposed administrative changes are offered to correct typographical errors and delete cycle specific footnotes which are no longer applicable. The nature of the changes precludes them from impacting previously analyzed accidents.

The proposed changes therefore do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2) Create the possibility of a new or different kind of accident from any accident previously evaluated because:

a. The proposed changes increase the STI and AOT for certain actuation instrumentation in the RPS, ECCS, Isolation, CRBF, RCIC, ATWS-RPT, EOC-RPT, Monitoring, and Feedwater/Main Turbine Trip systems. There are no changes in instrumentation configuration and function, and no instrumentation setpoints are changed.

b. The changes to the Feedwater/Main Turbine Trip LCO action statements allow the plant operators a maximum degree of operational flexibility, while maintaining the instrumentation and protection needed for terminating the feedwater controller failure transient. The single failure proof criterion of the level sensors is maintained, and the logic of the protective instrumentation is not compromised. The changes to the LCO action statements do not constitute a change to the facility or its operation as described in the Safety Analysis Report.

c. Deleting the requirement for surveilling the APRM Neutron Flux - High, Setdown functional unit in Operating Condition 1 does not degrade thermal margins. The margin accommodates the anticipated maneuvers associated with plant power ascension. During a plant shutdown, rod insertion maneuvers, recirculation flow reduction, and xenon build-in all contribute to negative reactivity insertion which precludes the degradation and violation of thermal margins. The functions of the APRMs required to be OPERABLE in Operational Condition 1 which are in effect remain to ensure that reactor core thermal margins are not compromised.

d. The conduct of neutron instrument functional tests in the plant mode or condition in which the trips are applicable eliminates unnecessary testing during normal plant operations. The expected result of the functional testing is to demonstrate the operability of the instruments. The failure of any single instrument channel will neither cause nor prevent either a reactor scram or a control rod block.

e. Including the performance of channel checks for the applicable analog instrumentation as part of the technical specifications transfers control of the required surveillances from procedure to the technical specifications, as appropriate. The administrative nature of this change does not alter the functions, setpoints, or configuration of the associated instrumentation.

f. The administrative nature of the changes prevents them from affecting the functions, setpoints, or configuration of the associated instrumentation from being affected by the changes.

The proposed changes do not create the possibility for an accident or malfunction of a different type than any previously evaluated in the UFSAR.

3) Involve a significant reduction in the margin of safety because:

a. Setpoints are based upon the drift occurring during an 18 month calibration interval. The bases in the Technical Specifications either do not discuss STI, or state "...one channel may be inoperable for brief intervals to conduct required surveillance." The proposed changes are bounded by the analyses of the topical reports. These analyses, which were prepared by GE and approved by the NRC, examined the effects of extending STI and AOT and found that the proposed changes would not involve a significant reduction in the margin of safety.

b. The proposed changes to the turbine trip LCO action statements do not change any of the settings of the Level 8 setpoints. The single failure criteria of the multiple level sensors which sense and detect the Level 8 setpoint remains intact. The LCO maintains the requirement that no single instrument failure will prevent the feedwater pump turbines and main turbine trip on a valid Level 8 signal. Scram trip signals from the turbine retain the design feature that a single failure will neither initiate nor impede the initiation of a reactor scram (trip).

c. The setting, function, and conditional requirements of the APRM Neutron Flux - High, Setdown function are not altered. This change serves to achieve consistency between two Technical Specifications Tables. This eliminates the need for surveilling a function in a mode which is not applicable. The functions of the APRMs required to be OPERABLE in Operational Condition 1 remain to ensure that reactor core thermal margins are not compromised.

d. The reference to 4.0.4 applicability will assist to ensure consistent interpretation of the technical specifications by the plant operators. This assists in ensuring that the plant is operated within technical specification limitations. This change does not affect trip instrumentation setpoints, and the scram function of the RPS is assured by the weekly functional testing of the Manual Scram.

e. Including the instrumentation channel checks as part of technical specification requirements provides an appropriately regimented method of controlling the conduct of the surveillances. None of the functions, setpoints, or configuration of the associated analog instrumentation is affected by this administrative change.

f. The administrative nature of the changes serves to provide more concise guidance to the plant operating staff, and as such do not impact the safety margin.

The proposed changes do not significantly reduce the margin of safety as defined in the basis for any Technical Specification.

Guidance has been provided in "Final Procedures and Standards on No Significant Hazards Considerations," Final Rule, 51 FR 7744, for the application of standards to license change requests for determination of the existence of significant hazards considerations. This document provides examples of amendments which are and are not considered likely to involve significant hazards considerations. This proposed amendments most closely fit the example of a change which may either result in some increase to the probability or consequences of a previously analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the applicable Standard Review Plan.

This proposed amendment does not involve a significant relaxation of the criteria used to establish safety limits, a significant relaxation of the bases for the limiting safety system settings or a significant relaxation of the bases for the limiting conditions for operations. Therefore, based on the guidance provided in the Federal Register and the criteria established in 10 CFR 50.92(c), the proposed change does not constitute a significant hazards consideration.

ATTACHMENT D

ENVIRONMENTAL ASSESSMENT STATEMENT APPLICABILITY REVIEW

Commonwealth Edison has evaluated the proposed amendment against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. It has been determined that the proposed changes meet the criteria for a categorical exclusion as provided under 10 CFR 51.22 (c)(9). This conclusion has been determined because the changes requested do not pose significant hazards consideration or do not involve a significant increase in the amounts, and no significant changes in the types, of any effluents that may be released offsite. Additionally, this request does not involve a significant increase in individual or cumulative occupational radiation exposure.

ATTACHMENT I

NON-PROPRIETARY VERSION OF THE GENERAL ELECTRIC TOPICAL REPORT
TECHNICAL SPECIFICATION IMPROVEMENT ANALYSIS
FOR THE REACTOR PROTECTION SYSTEM FOR LASALLE
COUNTY STATIONS, UNITS 1 AND 2

MDE-83-0485 REV. 3. DRF C71-00072-1, APRIL 1991

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MDE-83-0485-NP Rev 3
DRF C71-00072-2

APRIL 1991

GENERAL ELECTRIC COMPANY

TECHNICAL SPECIFICATION IMPROVEMENT ANALYSIS FOR THE REACTOR PROTECTION SYSTEM FOR LASALLE COUNTY STATION UNITS 1 AND 2

(THIS REPORT HAS BEEN PREPARED FOR COMMONWEALTH EDISON COMPANY THROUGH THE TECHNICAL SPECIFICATION IMPROVEMENT COMMITTEE OF THE BWR OWNERS' GROUP)

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TABLE OF CONTENTS

1. INTRODUCTION

This report extends the generic study of modifying the technical specification requirements of the Reactor Protection System (RPS) on a plant specific basis for LaSalle County Station (LSCS) Units 1 and 2. The generic study (Reference 1) provides a technical basis to modify the surveillance test frequencies and allowable out-of-service time of the RPS from the generic technical specifications. The generic study also provides additional analyses of various known different RPS configurations to support the application of the generic basis on a plant specific basis. The generic basis and the supporting analyses were utilized in this plant specific evaluation. The results of the plant specific evaluation for LSCS are presented herein.

This report represents the latest plant configuration of LSCS as of December 1990.

2. EVALUATION METHOD

The plant specific evaluation of the modification of the surveillance test frequencies and allowable out-of-service time of the RPS was performed in the following steps:

a. Gather plant specific information on the RPS from the Commonwealth Edison Company (CECo). The information includes the following:

   (1) Elementary Diagram of the RPS and related systems.
   (2) RPS description such as plant Final Safety Analysis Report (FSAR).
   (3) Technical specifications on the RPS.
   (4) RPS surveillance test procedures.

   The latest revision of Items 1,2, and 3 above were supplied by CECo. Item 4 above was provided by CECo in the form of a questionnaire identifying the differences between the procedure used in the generic evaluation and the procedure used at LSCS. Section I of the checklist in Appendix A was used to identify the data source of the plant specific information.

b. Construct the plant specific RPS configuration from the plant specific information. Questions "A" through "H" in Section II of the checklist were used for this process.

c. Compare the plant specific RPS configuration with the generic RPS configuration using the generic RPS elementary diagram, RPS description, technical specification requirements, and other generic inputs. Section III of the checklist was used for this process.

d. Classify the differences into three categories:

(1) Obvious items which have no effect on the reliability of the RPS. Examples of these "no effect" items are component name differences, symbol differences, and other minor nonfunctional differences. Disposition of the obvious "no effect" items does not require additional analysis.

(2) Potential differences which require considerable engineering judgment for disposition because of the functional differences. Examples of these potential differences are separate channels for manual scram as opposed to non-separate channel in the generic plant and dual redundant contacts per sensor relay in the applicable trip channels as opposed to a single set of contacts in the generic plant. The disposition of such items would require engineering assessment as shown in Appendix K of Reference 1.

(3) Potential differences which require additional analyses to evaluate the effect on the RPS reliability. Examples of such potential differences are using HFA relays as opposed to using both Potter and Brumfield relays and Agastat relays in the generic evaluation. Disposition of these items would require additional analyses to compare with the generic results. These analyses are documented in Reference 1.

e. Compile a list of plant specific differences of Category (2) and (3).

f. Assess the reliability effect of the differences identified in Step (e) on the generic results. The results of the assessment are documented in Section III of the checklist.

g. Document the results of the plant specific evaluation.

The above seven step process is documented in Appendix A of this report.

3. RESULTS OF RPS EVALUATION

The results of the plant specific evaluation of the RPS for LSCS are documented in Appendix A of this report. The results show that the RPS configuration of LSCS has the following differences which are classified Category (2) or (3):

* The term "generic model" means the RPS configuration used in the generic analysis (Reference 1).

[ ]

4. SUMMARY AND CONCLUSIONS

A plant specific evaluation of modifying the surveillance test frequencies and allowable out-of-service time of the RPS from the technical specifications of LSCS has been performed. The evaluation utilized the generic basis and the additional analyses documented in Reference 1. The results indicated that the RPS configuration for LSCS has several differences compared to the RPS configuration in the generic evaluation. These differences and the assessment of their effects on the RPS failure frequency are shown in Appendix A. The analysis reported in Reference 1 shows that these differences would not significantly affect the improvement in plant safety due to the changes in the technical specifications based on the generic analysis. Therefore, the generic analysis in Reference 1 is applicable to LSCS.

5. REFERENCES

(1) "Technical Specification Improvement Analyses for BWR Reactor Protection System," General Electric Company, NEDC-30851P-A, March 1988.

APPENDIX A

RPS EVALUATION CHECKLIST FOR LA SALLE COUNTY STATION UNITS 1 AND 2

Section I - RPS Plant Specific Data Source

Utility: Commonwealth Edison Company

Plant: LaSalle County Station Units 1 and 2

Source Number

1. RPS Elementary a) Unit 1 - 1E-1-4215AA through AM
   b) Unit 2 - 1E-2-4215AA through AM

2. RPS IED 732E152A, Rev. 6, 4 sh.

3. RPS MG Set Control System Elementary a) Unit 1 - 1E-1-4216AA, AB, AC
   b) Unit 2 - 1E-2-4216AA, AB, AC

4. RPS Interconnection Scheme Elementary 807E167TD, Rev. 5, 3 sh.

5. RPS Design Specification 22A3083AN, Rev. 10

6. FSAR Section 7.1.2 and 8.1.2, Rev. 6, April 1990

7. Technical Specifications Section 3/4.3 Amendment No. 75 (Unit 1)
   3/4.3 Amendment No. 59 (Unit 2)

8. Surveillance Test Procedure Checklist EDT BOA-8509

9. Others: EDT BOA-8537, Jim Marshall to Lynne Rash 03/07/85

Section I - RPS Plant Specific Data Source
Revision No. of RPS Elementary Drawings

	Unit 1	Unit 2
RPS Elementary	1E-1-4215AA Rev. F	1E-2-4215AA Rev. D
	1E-1-4215AB Rev. F	1E-2-4215AB Rev. D
	1E-1-4215AC Rev. AG	1E-2-4215AC Rev. AA
	1E-1-4215AD Rev. AJ	1E-2-4215AD Rev. AA
	1E-1-4215AE Rev. AM	1E-2-4215AE Rev. AE
	1E-1-4215AF Rev. AJ	1E-2-4215AF Rev. Y
	1E-1-4215AG Rev. F	1E-2-4215AG Rev. C
	1E-1-4215AH Rev. J	1E-2-4215AH Rev. K
	1E-1-4215AJ Rev. D	1E-2-4215AJ Rev. B
	1E-1-4215AK Rev. S	1E-2-4215AK Rev. L
	1E-1-4215AL Rev. N	1E-2-4215AL Rev. M
	1E-1-4215AM Rev. G	1E-2-4215AM Rev. E
	1E-1-4207ZB Rev. C	1E-2-4207ZB Rev. E
	1E-1-4232AW Rev. B	1E-2-4232AW Rev. A
	1E-1-4232AX Rev. B	1E-2-4232AX Rev. A
RPS MG Set Control System Elementary	1E-1-4216AA Rev. H	1E-2-4216AA Rev. K
	1E-1-4216AB Rev. C	1E-2-4216AB Rev. C
	1E-1-4216AC Rev. A	1E-2-4216AC Rev. A

Section II - RPS Configuration Data

A.	<u>RPS System</u>	<u>Data</u>	*Data <u>Source</u>
1.	Number of trip systems	2	(2,6)
2.	Number of logic channels per trip system		
	- For Automatic Scram	2	(1,2)
	- For Manual Scram	2	(1)
3.	Power supply source for each channel	MG Set	(2,6)
4.	Operation mode		
	- De-energize to trip	Yes	(1,6)
5.	Logic arrangement		
	- one-out-of-two twice	Yes	(6)
6.	Electrical Protection Assemblies (EPAs)	Yes	(6)
7.	Design requirement IEEE-279		(6)

* The numbers shown in the Data Source column refer to the documents listed in Section I.

Section II - RPS Configuration Data

B. RPS Sensors

1. Identify the type, total number, and number per RPS channel for the following RPS sensors.

	Type	Total Number	Number/ RPS Channel	Data Source
- APRM	Analog	6	2	(1,5)
- Turbine Stop Valve	Switch	8	2	(1,5)
- Turbine Control Valve	Switch	4	1	(1,5)
- MSIV Position	Switch	8	4	(1,5)
- MSL Radiation	Gamma Detector	4	1	(1,5)
- Level 8 (High Water Level)	N/A	N/A	N/A	N/A
- Level 3 (Low Water Level)	Analog	4	1	(1,5)
- SDV Level				
Type 1 (Analog)	Analog	4	1	(1,5)
Type 2 (Switch)	Switch	4	1	(1,5)
- High Reactor Pressure	Switch	4	1	(1,5)
- High Drywell Pressure	Switch	4	1	(1,5)
- Manual Trip	Switch	4	1	(1,5)
- Mode Switch Trip	Switch	1	1	(1,5)
- Low Condenser Vacuum	N/A	N/A	N/A	(1,5)
- Low Scram Air Header Pressure	N/A	N/A	N/A	(1,5)
- Low CRD Charging Water Header Pressure	Analog	4	1	(1)

Section II - RPS Configuration Data

		Data
B.	<u>RPS Sensors (Cont'd)</u>	<u>Source</u>
2.	Turbine Stop Valve closure logic arrangement Closure of 3 out of 4 valves initiates scram	(1)
3.	Turbine Stop Valve closure monitoring Position switches	(6)
4.	Turbine Control Valve fast closure monitoring Oil Pressure Switches	(6)
5.	MSIV closure logic arrangement Closure of 3 out of 4 steamlines initiates scram	(1)
6.	Diversity in SDV level sensors Yes	(1,5)
7.	Number of MSL 4	(6)
8.	List of available bypasses	(5)

Bypass	
- IRM Trip Bypass	No
- Noncoincident Neutron Monitoring System Trip Bypass	No
- RPV High Water Level RPS Trip Bypass	N/A
- Turbine Stop Valve RPS Trip Bypass	Yes
- Turbine Control Valve RPS Trip Bypass	Yes
- MSIV Closure RPS Trip Bypass	Yes
- SDV High Water Level Trip Bypass	Yes
- Reactor Mode Switch "Shutdown" mode Trip Bypass	No

Section II - RPS Configuration Data

C. Sensor Relays	Data	Data Source
1. Types of relays	GE Type HFA, HMA, CR2820 Agastat Type GP, EGP, E	(1)
2. Number of pairs of contacts per relay in the trip channel	2	(1)
3. List type of relay for each RPS sensor		(1)

	Potter & Brumfield	Agastat	HFA	GE
- APRM			x	
- Turbine Stop Valve			x	
- Turbine Control Valve			x	
- MSIV Position			x	
- MSL Radiation			x	x
- Level 3			x	
- SDV Level			x	
Type 1 (Analog)			x	
Type 2 (Switch)			x	
- High Reactor Pressure			x	
- High Drywell Pressure		x	x	
- Manual Trip			x	
- Mode Switch Trip		x	x	x
- Low Condenser Vacuum			N/A	
- Level 8			N/A	
- Low Scram Air Header Pressure			N/A	
- Low CRD Charging Water Header Pressure		x		

Section II - RPS Configuration Data

D.	Scram Contactors	Data	Data Source
1.	Type of scram contactors	GE Type CR105, CR205, and CR305	(1)
2.	Total number of scram contactors	8*	(1)
3.	Number of scram contactors per channel	2	(1)

* Manual scram channel shares the same scram contactors as its corresponding auto scram channel.

Section II - RPS Configuration Data

		Data	Data Source
E.	Air Pilot Solenoid Valves		
1.	Number of solenoid valves per control rod drive	2	(2)
2.	Number of solenoid operators per valve	1	(2)

Section II - RPS Configuration Data

F.	Backup Scram	Data	Data Source
1.	Type of scram contactors for Backup Scram Valves	GE Type CR105, CR205, and CR305	(1)
2.	Number of scram contactors per Backup Scram Valve	4	(1)
3.	Same RPS scram contactors are used to actuate Backup Scram Valves	Yes	(1)
4.	Operator mode - energized to trip	Yes	(1)
5.	Test requirement for Backup Scram Valves	Not specified in Tech. Spec.	(1)

Section II - RPS Configuration Data

		Data Source
G.	RPS Tech. Spec. Requirements	
1.	Calibration Frequency for LPRM At least once per 1000 Effective Full Power Hours	(7)
2.	Calibration frequency for trip units. Every refueling outage except for Reactor High Pressure and Drywell High Pressure trip units calibration done every 3 months	(7)
3.	Frequency of Logic System Functional Tests Every 18 months	(7)
4.	Allowable time to place an inoperable channel or trip system in the tripped conditions when the number of operable channels is less than the required minimum operable channels per trip system. 1 hour	(7)
5.	Exception to item 4. Restore to operable status within 2 hrs	(7)
6.	Allowable time to place a trip system in the tripped conditions when the number of operable channels is less than the required minimum operable channels for both trip systems. 1 hour	(7)
7.	Exception to item 6 due to surveillance test. May be in inoperable status up to 2 hrs two hours for surveillance	(7)
8.	Complete the Table on the following page.	

REACTOR PROTECTION SYSTEM INSTRUMENTATION REQUIREMENTS

Functional Unit	Channel Check		Channel Functional Test		Channel Calibration		Minimum Operable Channels Per Trip System	
	Generic Model	Plant Specific	Generic Model	Plant Specific	Generic Model	Plant Specific	Generic Model	Plant Specific
1. Average Power Range Monitor:								
a. Flow Biased Simulated Thermal Power - High	S,D	S,D	S/U,W	S/U,W	W,SA,R	W,SA,R	3	2
b. Neutron Flux - High	S	S	S/U,W	S/U,W	W,SA	W,SA	3	2
c. Inoperative	N/A	N/A	W	W	N/A	N/A	3	2
2. Reactor Vessel Steam Dome Pressure - High	S	N/A	M	M	R	Q	2	2
3. Reactor Vessel Water Level - Low, Level 3	S	S	M	M	R	R	2	2
4. Reactor Vessel Water Level - High, Level 8	S	N/A	M	N/A	R	N/A	2	N/A
5. Main Steam Line Isolation Valve - Closure	N/A	N/A	M	M	R	R	4	4
6. Main Steam Line Radiation - High	S	S	M	M	R	R	2	2
7. Drywell Pressure - High	S	N/A	M	M	R	Q	2	2
8. Main Condenser Vacuum - Low	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A

REACTOR PROTECTION SYSTEM INSTRUMENTATION REQUIREMENTS

Functional Unit	Channel Check		Channel Functional Test		Channel Calibration		Minimum Operable Channels Per Trip System	
	Generic Model	Plant Specific	Generic Model	Plant Specific	Generic Model	Plant Specific	Generic Model	Plant Specific
9. Scram Discharge Volume Water Level - High								
Type 1 - Analog	S	N/A	M	M	R	R	2	2
Type 2 - Switch	N/A	N/A	M	M	R	R	2	2
10. Turbine Stop Valve - Closure	N/A	N/A	M	M	R	R	4	4
11. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	N/A	N/A	M	M	R	R	2	2
12. Reactor Mode Switch Shutdown Position	N/A	N/A	R	R	N/A	N/A	2	1
13. Manual Scram	N/A	N/A	M	M	N/A	N/A	2	1
14. Low Air Header Scram Pressure - Low	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A

REACTOR PROTECTION SYSTEM INSTRUMENTATION REQUIREMENTS

Functional Unit	Channel Check		Channel Functional Test		Channel Calibration		Minimum Operable Channels Per Trip System	
	Generic Model	Plant Specific	Generic Model	Plant Specific	Generic Model	Plant Specific	Generic Model	Plant Specific
15. Control Rod Drive:								
a. Charging Water Header Pressure - Low	N/A	N/A	N/A	M	N/A	R	N/A	2
b. Delay Timer	N/A	N/A	N/A	M	N/A	R	N/A	2

S = Shift
D = Daily
W = Weekly

M = Monthly
Q = Quarterly
R = Refueling

S/U = Startup
N/A = Not Applicable
SA = Semi Annual

Section II - RPS Configuration Data

		Data Source
H.	RPS Surveillance Tests Procedure	

1. The following components are all tested as part of an individual channel functional test: (8)

   a. Individual channel sensor(s), e.g., Transmitters and Trip Units, switches, flux or radiation sensors.

   b. Associated logic relay(s)

   c. Associated scram contactors

   List any plant specific differences from the above.

   RESPONSE

   We do not verify bypassed during functional, only during logic test. Also we don't functionally test rad detectors or flux detectors themselves. (Just electronic trip units).

2. When an individual sensor channel is in test or repair, is associated logic channel tripped or is the sensor channel jumpered? State which of the two conditions applies to your plant. If any other condition exists in your plant, describe. (8)

   RESPONSE

   The instrument, when in test, is not functional (i.e. valved out, out of operation, etc.). However the channel may be tripped (1/2 scram) or untripped throughout the course of the surveillance. In general I guess jumpered would apply although we do not jumper sensors during surveillance. During surveillance/repair, the Tech. Spec. time clock would apply and require tripping the channel within 2 hours (surveillance) or 1 hour (repair). Up to this time the channel may or may not be tripped.

Section II - RPS Configuration Data (Cont'd)

H. RPS Surveillance Tests Procedure (Cont'd) — Data Source

3. For those plants which do not place individual channels in a tripped condition during test or repair, it is assumed in the GE analysis that only the individual sensor and associated logic relay is placed in an inoperable condition during test or repair of the individual channel. If this assumption is not true for your plant, list the components (from sensor to scram contactors) which are placed in inoperable condition during test or repair. (8)

RESPONSE

This is true, only the channel is affected.

4. The following number of individual scram contactor actuations are assumed in the GE analyses for each channel functional test: (8)

a. APRM channel functional tests - 2 actuations per scram contactor pair in each trip logic channel.

b. MSIV closure channel function tests - 4 actuations per scram contactor pair in each trip logic channel.

c. Other channel functional tests - 1 actuation per scram contactor pair in each trip logic channel.

List any differences from the above for your specific plant.

RESPONSE

4.a. - LSCS - 4/contactor pair

4.b. - True - 4

4.c. - Normally true

Section II - RPS Configuration Data (Cont'd)

		Data Source
H.	RPS Surveillance Tests Procedure (Cont'd)	
5.	Do plant procedures allow simultaneous inoperable conditions (failed condition) of diverse sensors in a given logic channel?	(8)

RESPONSE

Yes within the limit of the Tech. Spec. time clocks described above.

Section III - Assessed Reliability Effect of RPS Configuration Differences

	BWR Generic Model	Plant Specific Difference	Assessed Reliability Effect
A.	RPS System		
1.	Generic model has two trip systems.	No plant specific difference	
2.	Generic model has two logic channels per trip system for automatic scram.	No plant specific difference	
3.	During operation, the trip systems are energized and trip when de-energized.	No plant specific difference	
4.	The RPS logic is one-out-of-two twice, i.e., one out of two logic channels will trip an individual system and trip of both systems is required for scram.	No plant specific difference	
5.	Generic model has Electrical Protection Assemblies (EPAs).	No plant specific difference	
6.	Each RPS channel can be manually tripped from the Control Room using the manual scram circuits.	No plant specific difference	
7.	Generic model has MG set power supply.	No plant specific difference	

Section III - Assessed Reliability Effect of RPS Configuration Differences

	BWR Generic Model	Plant Specific Difference	Assessed Reliability Effect
B.	Sensors		
1.	Generic model has Analog Trip Unit/ Transmitter for pressure sensors.	Switches used for pressure sensors.	
1A.	Generic model has Analog Trip Unit/ Transmitter for level sensors.	No plant specific difference	
2.	Minimum number of sensors is one per RPS channel for each scram variable.	No plant specific difference	
3.	Generic model has eight APRM monitors with two for RPS channel.	Six APRM monitors w/ two APRM shared by two channels.	
4.	Stop Valve Closure trip logic is a reduced three-of-four required for trip.	No plant specific difference	
5.	Stop Valve Closure is monitored by limit switches.	No plant specific difference	

Section III - Assessed Reliability Effect of RPS Configuration Differences

	BWR Generic Model	Plant Specific Difference	Assessed Reliability Effect
B.	Sensors (Cont'd)		
6.	Turbine Control Valve fast closure is monitored by control oil pressure.	No plant specific Difference	
7.	MSIV closure trip logic requires isolation of three out of four steam lines to scram.	No plant specific difference	
8.	Generic model has a level 8 (High Reactor Water Level) Trip.	No Level 8 trip	No significant effect as demonstrated by analysis in Reference 1.
9.	Generic model has diverse Scram Discharge Volume (SDV) level sensors.	No plant specific difference	
10.	Generic model has 4 main steamlines.	No plant specific difference	
11.	Generic model does not have a direct scram on low condenser vacuum.	No plant specific difference	
12.	Generic model does not have a direct scram on low air header pressure.	No plant specific difference	
13.	Generic model does not address CRD Charging Water Header Pressure - Low scram signal.	CRD Charging Water Header Pressure - Low scram signal	[ ]

Section III - Assessed Reliability Effect of RPS Configuration Differences

	BWR Generic Model	Plant Specific Difference	Assessed Reliability Effect
C.	Sensor Relays		
1.	For all transients there are at least two scram variables with different type logic relays (either Agastat or Potter & Brumfield).	GE type HFA, HMA, & CR2820, and Agastat type GP, EGP, & E relays used for all scram variables	
2.	Each sensor relay has a single pair of contacts in the applicable trip channel.	Two sets of contact pair per sensor relay	
D.	Scram Contactors		
1.	All Scram Contactors are one type (GE Type CR105).	Scram Contactors are GE Type CR105, CR205, and CR305	
2.	Eight scram contactors (two per RPS channel) perform the trip function.	No plant specific difference (two per RPS channel)	
E.	Air Pilot Solenoid Valves		
1.	Generic model has dual solenoid operators for each individual HCU air pilot valve. De-energizing both solenoids results in a scram of the individual control rod.	Two HCU valves with single solenoid operators. Tripping of both valves is required for individual control rod scram.	

Section III - Assessed Reliability Effect of RPS Configuration Differences

	BWR Generic Model	Plant Specific Difference	Assessed Reliability Effect
F.	Backup Scram		
1.	Actuation of backup scram valves are controlled by same output scram contactors as RPS	No plant specific difference	
2.	Trip logic for backup scram valves is an energized to trip versus de-energized to trip for individual HCU air pilot valves.	No plant specific difference	
3.	Backup scram valves are tested during shutdown at least once per 18 months.	No requirement data available	

Section III - Assessed Reliability Effect of RPS Configuration Differences

BWR Generic Model	Plant Specific Difference	Assessed Reliability Effect
G. Technical Specifications and Surveillance Test Procedure:		
1. Generic model uses BWR6 Standard Technical Specifications which requires: Allowable out-of-service time: 1 hr Test time: 2 hrs Test frequency: 1W for APRM 1M for others Calibration frequency: 1M for trip units R for transmitters	See Section II.G of this Appendix for plant specific differences.	
2. Generic model assumes 2 actuations per scram contactor pair in each trip logic for the MSIV closure channel functional test, and one actuation for the other scram variables. This leads to 272 total actuations of each scram contactor per year.	Four actuations per scram contactor pair for both APRM and MSIV Closure channel functional test and one actuation for the other scram variables. This leads to 352 total actuations of each scram contactor per year.	

ATTACHMENT J

NON-PROPRIETARY VERSION OF THE GENERAL ELECTRIC TOPICAL REPORT
TECHNICAL SPECIFICATION IMPROVEMENT
ANALYSIS FOR THE EMERGENCY CORE COOLING
SYSTEM ACTUATION INSTRUMENTATION FOR
LASALLE COUNTY STATION, UNITS 1 AND 2

RE-025, REV 1, DRF C71-00072-1, APRIL 1991

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GENERAL ELECTRIC COMPANY

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IMPORTANT NOTICE REGARDING CONTENTS OF THIS REPORT

Please Read Carefully

TABLE OF CONTENTS

1. INTRODUCTION

This report extends the generic study of modifying the technical specification requirements of the emergency core cooling system (ECCS) on a plant specific basis for LaSalle County Station, Units 1 and 2, BWR 5s. The generic study (References 1 and 2) provides a technical basis to modify the surveillance test intervals and allowable out-of-service times of the ECCS actuation instrumentation from those of the generic technical specifications. The generic study also provides additional analyses of various known different ECCS configurations to support the application of the generic basis on a plant specific basis. The generic basis and the supporting analyses were utilized in this plant specific evaluation. The results of the plant specific evaluation for LaSalle are presented herein.

This report represents the latest plant configuration of LSCS as of December 1990.

2. EVALUATION METHOD

The plant specific evaluation of the modification of the surveillance test frequencies and allowable out-of-service times of the ECCS actuation instrumentation was performed in the following steps:

a. Gather plant specific information on the ECCS from Commonwealth Edison Company (CECO). The information includes the following:

   (1) Piping and Instrumentation Diagrams (P&IDs) of ECCS, reactor core isolation cooling (RCIC) system, emergency service water systems, and air systems to ADS valves.
   (2) Elementary Diagrams of the ECCS, RCIC, and related systems.
   (3) ECCS, RCIC and electric power distribution system descriptions such as those in the plant Final Safety Analysis Report (FSAR).
   (4) Technical specifications on the ECCS, RCIC, the suppression chamber, and the electrical systems.
   (5) Information on ECCS surveillance test procedures.
   (6) Dependency matrices showing dependence of ECCS and RCIC systems on support systems and on actuation instrumentation.
   (7) Available data on actuation instrumentation failures.

   The latest revisions of the above items were supplied by CECO. Section I of the checklist in Appendix A was used to identify the data source of the plant specific information.

b. Construct the plant specific ECCS configuration from the plant specific information. Sections "A" through "E" in Section II of the Appendix A checklist was used for this process.

c. Compare the plant specific ECCS configuration with the generic ECCS configuration using the generic ECCS fault trees, ECCS description, technical specification requirements, and other generic inputs. Section III of the checklist was used for this process.

d. Classify the differences in ECCS system design, in support systems, and in instrumentation, into three categories:

(1) Differences which obviously have no negative effect on the reliability of the ECCS. Examples of these "no effect" items are component name differences, symbol differences, and other minor non-functional differences. Other effects not requiring analysis are those in which the specific plant has greater redundancy than the generic model. Disposition of the items with obviously no negative effect is done with "no analysis required".

(2) Differences which require engineering judgement for disposition because of the functional differences. Examples of these differences are the use of shared room cooling systems in a specific plant compared with individual room cooling systems in the generic plant. The disposition of such items would require engineering assessment in a "simple study" as shown in Appendix F of Reference 2.

(3) Differences which require additional analyses to evaluate the effect on the ECCS reliability. Examples of such differences are the use of two diesel generators and two electrical systems in a specific plant compared with a larger number of diesel generators and electrical systems in the generic evaluation. Disposition of these items would require additional analyses ("Modify fault trees and perform analysis.") to compare with the generic results. These analysis are documented in Reference 2.

e. Compile a list of plant specific differences of Categories (2) and (3).

f. Assess the reliability effect of the differences identified in Step (e) on the generic results. The results of the assessment are documented in Section III of the checklist, Appendix A.

g. Document the results of the plant specific evaluation.

The above seven step process is documented in Appendix A of this report.

3. RESULTS OF ECCS EVALUATION

The results of the plant specific evaluation of the ECCS for LSCS are documented in Appendix A of this report. The results show that the ECCS and support systems configuration of LSCS has three differences from the BWR 5/6 generic model* which are classified Category (3), and one which is in Category (2).

3.1 Detailed Analyses

The LSCS differences in Category (3), requiring detailed analysis, are as follows:

a. The generic model has the LPCI and LPCS injection valve permissive signal from RPV pressure, using one-out-of-two-twice logic; LSCS uses 1/2 RPV pressure signals plus 1 valve pressure signal for this permissive.

b. The generic model has no ADS inhibit switch, LSCS has an ADS inhibit switch.

c. Injection valves in the generic model are stroke tested quarterly, at LSCS the valve stroke test is performed at refueling, which could be as long as 18 months.

---

* The term "generic model" means the ECCS configuration used in the generic analysis.

3.2 Simple Studies

The LSCS difference in Category (2), requiring a simple study, is as follows:

a. The generic model has one diesel generator for each electrical division, LSCS had two dedicated DGs at each unit, with a fifth DG shared between the two units.

"The diesel-generator sets have ample capacity to supply all power required for the safe shutdown of both units in the event of a total loss of offsite power. Ample capacity is provided for the condition in which one unit may be involved in a loss-of-coolant accident while the remaining unit is being shut down without loss of coolant, as well as for the condition in which both units are concurrently being shut down without loss-of-coolant accidents."

4. SUMMARY AND CONCLUSIONS

A plant specific evaluation of modifying the surveillance test intervals and allowable out-of-service times of the ECCS from the technical specifications of LSCS has been performed. The evaluation utilized the plant specific information supplied by CECO and the generic basis and the additional analyses documented in References 1 and 2.

The results indicate that the ECCS configuration for LSCS is similar to the ECCS configuration in the generic evaluation, with four significant differences. The differences between LSCS and the generic model have been modeled by envelope cases 5A and 5C of Reference 2, plus one simple study, which show that the proposed changes to ECCS actuation instrumentation Technical Specifications would meet the acceptance criteria in Reference 2. Therefore, the generic basis in References 1 and 2 is applicable to LSCS.

5. REFERENCES

(1) D. B. Atcheson, et al., "BWR Owner's Group Technical Specification Improvement Methodology (with Demonstration for BWR ECCS Actuation Instrumentation) Part 1", General Electric Company, NEDC-30936P-A, December 1988.

(2) D. B. Atcheson, et al., "BWR Owner's Group Technical Specification Improvement Methodology (with Demonstration for BWR ECCS Actuation Instrumentation) Part 2", General Electric Company, NEDC-30936P-A, December 1988.

APPENDIX A

ECCS ACTUATION INSTRUMENTATION EVALUATION FOR LASALLE COUNTY STATION, UNITS 1 AND 2

Section I - ECCS Plant Specific Data Source

Utility: Commonwealth Edisor Co.
Plant: LaSalle County Station, Units 1 & 2

Source Number	
1.	ECCS and RCIC P&IDs
2.	Emergency Service Water P&IDs
3.	Electrical Drawings
4.	Instrumentation Logic Diagrams
5.	ECCS Fault Trees
6.	Final Safety Analysis Report
7.	Technical Specifications
8.	Other Drawings
9.	Dependency Matrices
10.	Failure Data
11.	Test Procedure Questionnaire
12.	Telephone Call Records
13.	NEDC-30936P-A, Part 1

Section II - ECCS Configuration Data

A. ECCS System

	LSCS	Generic BWR 5/6	Difference (Y/N)	Data* Source
1. Number of:				
- LPCS Pumps/Loops	1/1	1/1	N	1
- LPCI Pumps	3	3	N	1
- ADS Valves	7	8	Y	1
- HPCS Pumps	1	1	N	1
2. Needed for Success, Number of:				
- LPCS Pumps/Loops	1/1	1/1	N	6
- LPCI Pumps	1	1	N	13
- ADS Valves	3	3	N	13
3. Number of:				
- Diesel Generators	3**	3	Y	3
- Electrical Divisions	3	3	N	3

* The numbers shown in the Data Source column refer to the documents listed in Section 1.

** 2 Dedicated diesel generators per unit, 1 DG is shared between Units 1 & 2

Section II - ECCS Configuration Data

B. SUPPORT SYSTEM DEPENDENCIES

The dependencies each front line ECCS system has on the listed support subsystems for both the generic and specific plant are shown.

FRONT LINE SYSTEMS

SUPPORT SUBSYSTEMS	LPCI A	LPCI B	LPCI C	LPCS	ADS A	ADS B	RCIC	HPCS	DIESELS A	DIESELS B	DIESELS C
OFFSITE AC POWER	X	X	X	X			X	X			
ONSITE AC POWER											
DIVISION 1	X			X	X		X				
DIVISION 2		X	X			X	X				
DIVISION 3								X			
ONSITE DC POWER											
DIVISION 1	X			X	X		X		X		
DIVISION 2		X	X			X	X			X	
DIVISION 3								X			X
SERVICE WATER											
EMERGENCY LOOP A	X			X			X		X*		
EMERGENCY LOOP B		X	X							X	
EMERGENCY LOOP C								X			X
WATER SUPPLY											
CONDENSATE TANK							X	X			
SUPPRESSION POOL	X	X	X	X			X	X			
AIR											
CONTROL DIV 1					X						
CONTROL DIV 2						X					
CONTAIN. INSTR. DIV 1					X						
CONTAIN. INSTR. DIV 2						X					
ROOM COOLING											
LPCI	X	X	X								
LPCS				X							
RCIC							X				
HPCS								X			
DIESELS									X	X	X

X = IN BOTH GENERIC AND SPECIFIC BWR 5/6s
G = ONLY IN GENERIC BWR 5/6
S = ONLY IN SPECIFIC BWR 5/6
* = DG IS SHARED BETWEEEN UNITS 1 & 2

Section II - ECCS Configuration Data

C. INSTRUMENTATION DEPENDENCIES

The dependencies each front line ECCS system has on the listed actuation instrumentation for the generic and specific plants are shown.

FRONT LINE SYSTEMS

ACTUATION INSTRUMENTATION	LPCI A	LPCI B	LPCI C	LPCS	ADS A	ADS B	RCIC	HPCS
RPV WATER LEVEL 1 (LOW LOW LOW)								
B21-N707 A/C	X			X	X			
B21-N707 B/D		X	X			X		
RPV WATER LEVEL 2 (LOW LOW)								
B21-N710 A/B/C/D							X	
B21-N706 A/B/C/D								X
RPV WATER LEVEL 3 (LOW)								
B21-N708A					X			
B21-N708B						X		
RPV WATER LEVEL 8 (HIGH)								
B21-N709 A/B								X
B21-N705 A/B							X	
RPV PRESSURE LOW								
B21-N413 A/C	X			X				
B21-N413 B/D		X	X					
N698 A/E	G			G				
N698 B/F		G	G					
E21-N413				S				
E12-N413A	S							
E12-N413B		S						
E12-N413C			S					

X = IN BOTH GENERIC AND SPECIFIC BWR 5/6s
G = ONLY IN GENERIC BWR 5/6
S = ONLY IN SPECIFIC BWR 5/6

Section II - ECCS Configuration Data

C. INSTRUMENTATION DEPENDENCIES (Continued)

The dependencies each front line ECCS system has on the listed actuation instrumentation for the generic and specific plants are shown.

FRONT LINE SYSTEMS

ACTUATION INSTRUMENTATION	LPCI A	LPCI B	LPCI C	LPCS	ADS A	ADS B	RCIC	HPCS
DRYWELL PRESSURE HIGH								
B21-N048 A/C	X			X	X			
B21-N048 B/D		X	X			X		
LPCI PUMP DISCHARGE PRESSURE HIGH								
E12-N016 A/N019 A					X			
E12-N016 B/C						X		
E12-N019 B/C						X		
LPCS PUMP DISCHARGE PRESSURE HIGH								
E12-N001/N009					X			
ADS TIMER								
K35A					X			
K35B						X		
DRYWELL PRESSURE BYPASS TIMER								
K36A/K37A					X			
K36B/K37B						X		
ADS INHIBIT SWITCH								
S26A					S			
S26B						S		
MANUAL INITIATION SWITCH (1/LOOP)	X	X	X	X	X	X	X	X

X = IN BOTH GENERIC AND SPECIFIC BWR 5/6s
G = ONLY IN GENERIC BWR 5/6
S = ONLY IN SPECIFIC BWR 5/6

Section II - ECCS Configuration Data

C. INSTRUMENTATION DEPENDENCIES (Continued)

The dependencies each front line ECCS system has on the listed actuation instrumentation for the generic and specific plants.

FRONT LINE SYSTEMS

RELATED **NON-ACTUATION INSTRUMENTATION**	LPCI A	LPCI B	LPCI C	LPCS	ADS A	ADS B	RCIC	HPCS
LPCI/LPCS PUMP DISCHARGE FLOW LOW								
E12-N010 AA	X							
BA		X						
CA			X					
E21-N004				X				
CST LEVEL LOW								
E51-N035 A/B							X	
E22-N001 A/B								X
SUPPRESSION POOL WATER LEVEL HIGH								
N655 C/G							G	
E22-N002 A/B								X

X = IN BOTH GENERIC AND SPECIFIC BWR 5/6s
G = ONLY IN GENERIC BWR 5/6
S = ONLY IN SPECIFIC BWR 5/6

Section II - ECCS Configuration Data

D. Minimum Number of Sensors, Channels, or Components for Failure, LS 1 & 2

A: = MINIMUM NUMBER SENSOR FAILURES REQUIRED TO FAIL TRIP FUNCTION*
B: = MINIMUM NUMBER SENSOR FAILURES REQUIRED TO FAIL FUNCTION - TOTAL
C: = MINIMUM NUMBER OF SENSOR TYPES REQUIRED TO FAIL FUNCTION

TRIP FUNCTION	A	B	C	DIFFERENT FROM GENERIC (Y/N) B	DIFFERENT FROM GENERIC (Y/N) C
LPCS PUMP INITIATION	1 RPV WATER LEVEL 1 (LOW LOW LOW) AND 1 DRYWELL PRESSURE	2	2	N	N
LPCS INJ VALVE	1 RPV LOW PRESSURE	1	1	Y	N
LPCI PUMP INITIATION	1 RPV WATER LEVEL 1 AND 1 DRYWELL PRESSURE	2	2	N	N
LPCI INJ VALVES	3 RPV LOW PRESSURE	3	1	N	N
ADS INITIATION	2 RPV WATER LEVEL 1 OR 2 RPV WATER LEVEL 3 (LOW), OR 2 PUMP DISCH PRESS, OR 2 DRYWELL PRESS	2	1	N	N
ADS TIME DELAY	2 TIMERS	2	1	N	N
HPCS INITIATION	2 RPV LEVEL 2 (LOW LOW) AND 2 DRYWELL PRESSURE	4	2	N	N
HPCS LEVEL 8	2 RPV LEVEL 8 (HIGH)**	2	1	N	N
HPCS INJ VALVE	2 RPV LEVEL 2 AND 2 DRYWELL PRESSURE	4	2	N	N
HPCS WATER SOURCE	2 CST LEVEL AND 2 SUPPRESSION POOL LEVEL	4	2	Y	N
RCIC INITIATION	2 RPV LEVEL 2	2	1	N	N
RCIC LEVEL 8	2 RPV LEVEL 8**	2	1	N	N
RCIC WATER SOURCE	2 CST LEVEL	2	1	N	N
RCIC INJ VALVE	2 RPV LEVEL 2	2	1	N	N

* Based on data sources 4 & 6.
** For Level 8, trip function is false isolation of system.

Section II - ECCS Configuration Data

E. ECCS Instrumentation and related subsystems Surveillance Requirements*

	SURVEILLANCE REQUIREMENTS** GENERIC 5/6	LSCS	DIFFERENCE (Y/N)
CORE SPRAY SYSTEM			
REACTOR WATER LEVEL 1 (LOW LOW LOW)	M	M	N
DRYWELL PRESSURE HIGH	M	M	N
REACTOR PRESSURE LOW	M	M	N
MANUAL INITIATION	R	R	N
LPCI			
REACTOR WATER LEVEL 1	M	M	N
DRYWELL PRESSURE HIGH	M	M	N
REACTOR PRESSURE LOW	M	M	N
PUMP START TIME DELAY RELAY	M	M	N
INJECTION VALVE DIFFERENTIAL PRESSURE LOW	M	N/A	Y
MANUAL INITIATION	R	R	N
HPCS			
REACTOR WATER LEVEL 2 (LOW LOW)	M	M	N
DRYWELL PRESSURE HIGH	M	M	N
CST LEVEL LOW	M	M	N
SUPPRESSION POOL LEVEL HIGH	M	M	N
REACTOR WATER LEVEL 8	M	M	N
MANUAL INITIATION	R	R	N
RCIC			
REACTOR WATER LEVEL 2 (LOW LOW)	M	M	N
REACTOR WATER LEVEL 8	M	M	N
MANUAL INITIATION	R	R	N
ADS			
REACTOR WATER LEVEL 1	M	M	N
DRYWELL PRESSURE HIGH	M	M	N
ADS TIMER	M	M	N
CORE SPRAY PUMP DISCHARGE PRESSURE	M	M	N
LPCI PUMP DISCHARGE PRESSURE	M	M	N
REACTOR WATER LEVEL 3 (LOW)	M	[illegible]	N
MANUAL INITIATION	R	[illegible]	N
ADS DRYWELL PRESSURE BYPASS TIMER	M	M	N
ADS INHIBIT SWITCH	N/A	R	Y

---

* Based on Technical Specifications, data source No. 7.

** M - MONTHLY, W - WEEKLY, R - REFUELING, Q - QUARTERLY
CSD - COLD SHUT DOWN

Section II - ECCS Configuration Data

E. ECCS Instrumentation and related subsystems Surveillance Requirements* (Continued)

	SURVEILLANCE REQUIREMENTS** GENERIC 5/6	LSCS	DIFFERENCE (Y/N)
INJECTION VALVE STROKE TEST	Q	CSD/R	Y
DIESEL GENERATOR	M	M	N
ELECTRIC POWER			
ESSENTIAL AC	W	W	N
ESSENTIAL DC	W	W	N
ESSENTIAL AC BUSSES	W	W	N

* Based on Technical Specifications, data source No. 7.

** M = MONTHLY, W = WEEKLY, R = REFUELING, Q = QUARTERLY
CSD = COLD SHUT DOWN

Section III - ECCS Configuration Differences Classification
(LaSalle County Station)

BWR 5/6 Generic Model	Plant Specific Difference	Classification (Justification if Insignificant)
A. ECCS System Differences		
1. RC'C and HPCS have common portion of supply header from CST.	LSCS has independent supply lines from CST for RCIC and HPCS.	
2. Generic has 8 ADS valves.	LSCS has 7 ADS valves.	
B. Support System Differences		
1. Onsite AC power has 3 independent divisions.	Each unit has 2 independent divisions, Divs 2 & 3. Div 1 for each unit is from the shared diesel generator.	
C. Instrumentation and Procedures Differences		
1. Opening of LPCS & LPCI injection valves has RPV pressure permissive signal from 1/2-twice logic.	Opening of low pressure system injection valves requires one valve pressure signal plus 1/2 RPV pressure signals.	
2. Containment spray signal could prevent LPCI operation.	No containment spray signal interlock.	
3. Injection valves are stroke tested quarterly.	Injection valve stroke tests are performed at cold shutdown.	
4. No ADS inhibit switch.	ADS has inhibit switch.	

Section III - ECCS Configuration Differences Classification
(LaSalle County Station)

BWR 5/6 Generic Model	Plant Specific Cifference	Classification (Justification if Insignificant)
C. Instrumentation and Procedures Differences (Cont'd)		
5. Analog trip units are used.	LSCS uses process switches and analog trip units in sensor channels.	[ ]

ATTACHMENT K

GENERAL ELECTRIC LETTER TO R.H. MIROCHUA
(COMMONWEALTH EDISON COMPANY), TECHNICAL
SPECIFICATION IMPROVEMENT FOR BWR
INSTRUMENTATION TRANSMITTAL OF DELIVERABLES
LASALLE COUNTY STATION

EBO-90-246, MAY 1, 1991

GE Nuclear Energy

EBO-90-246

General Electric Company
2311 West 22nd St., Suite 201, Oak Brook, IL 60521
708 573-3929

May 1, 1991

Mr. R. H. Mirochna
Commonwealth Edison Company
Executive Towers West III
1400 Opus Place, Suite 400
Downers Grove, IL 60515

SUBJECT: TECHNICAL SPECIFICATION IMPROVEMENT FOR BWR INSTRUMENTATION
TRANSMITTAL OF DELIVERABLES
LA SALLE COUNTY STATION

References:
1. CECo Purchase Order 328658, Release No. NU-77, dated November 13, 1990.
2. GE Letter No. EBO-90-414, Same Subject, Budgetary Estimate No. 295-1BFH4-HA0-90, W. Arndt to R. Mirochna, dated October 5, 1990.

Dear Mr. Mirochna:

Enclosed please find the following items which complete GE's contracted scope of work of References 1 and 2:

1. Report MDE-83-0485, Revision 3, "Technical Specification Improvement Analysis for the Reactor Protection System for LaSalle County Station Units 1 and 2", dated April 1991.
2. General Electric Company Affidavit for Report MDE-83-0485.
3. Report RE-025, Revision 1, "Technical Specification Improvement Analysis for the Emergency Core Cooling System Actuation Instrumentation for LaSalle County Station, Units 1 and 2", dated April 1991.
4. General Electric Company Affidavit for Report RE-025.
5. A draft submittal letter to the NRC.
6. Copy of GE Letter No. OG9-1219-32D, Clarification of Limerick 1 & 2 Proposed Technical Specification Changes Common to Reactor Protection System or ECCS Actuation Instrumentation, W. Sullivan to BWROG Technical Specification Committee, dated December 22, 1989.
7. Copy of GE Letter No. OG90-319-32D, Clarification of Technical Specification Changes Given in ECCS Actuation Instrumentation Analysis, W. Sullivan and J. Klapproth to M. Wohl, dated March 22, 1990.

Report MDE-83-0485, Revision 3, completes Reference 2 scope of work Item 1. Report RE-025, Revision 1, completes Reference 2 scope of work Item 2.

To complete the Reference 2 scope of work Items 3 through 5, GE has verified that the conclusions of the Licensing Topical Reports of scope of work Items 3 through 5 are still applicable to LaSalle County Station and that plant specific reports are not necessary. The draft NRC submittal letter contains the results of this verification and is provided in a format which will facilitate CECo's amendment application. The draft NRC submittal letter is consistent with amendment applications submitted by other plants.

In addition to referencing scope of work Items 1 through 5, the draft NRC submittal letter also addresses the following specific items:

1. End-of-Cycle Recirculation Pump Trip (EOC-RPT) System.
2. RCIC System.
3. Clarification of on the intent of technical specification mark-up for ECCS Actuation Instrumentation.
4. Clarification of RPS Limited Condition of Operation (LCO).

This document transmittal to CECo fulfills all GE's commitments and provides all deliverables for the Reference 1 CECo Purchase Order.

GE thanks you for the opportunity to perform these services for CECo. If you have any questions, please call me, or Ron Ninomiya at (408) 925-2077, at your convenience.

Sincerely,

William D. Arndt
Senior Customer Service Engineer
(708) 573-3964

Attachments

cc:	CECo	GE
	J. W. Gieseker w/o att.	J. C. Elliott w/o att.
	D. E. Lockwood	G. L. Hayes w/o att.
	E. L. Seckinger	J. E. Kusky
	J. D. Williams w/o att.	R. B. Ninomiya w/o att.
	Chron System	File: 4.Z12.0

DRAFT

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: LaSalle County Station, Units 1 and 2
Technical Specifications Change Request

Dear Sir:

Commonwealth Edison Company hereby submits Technical Specifications Change Request No. __________ in accordance with 10 CFR 50.90, requesting an amendment to the Technical Specifications (TS) (Appendix A) of Operating License Nos. __________ and __________. Information supporting this Change Request is contained in Attachment 1 to this letter. Attachment 2 provides a list of references used to justify this Change Request, and Attachment 3 provides the proposed TS change pages.

This submittal requests changes to the TS to extend surveillance test intervals and allowable out-of-service items for instrumentation supporting the Reactor Protection System (RPS), Emergency Core Cooling System (ECCS), and Isolation Actuation including common instrumentation.

This letter also submits the following documents which provide additional information supporting this Change Request, as enclosures.

a) Enclosure 1 - General Electric (GE) Document No. OG9-1219-32D, letter W. P. Sullivan, GE, to Boiling Water Reactor Owners' Group (BWROG) Technical Specification Committee, dated December 22, 1989: "Clarification of Limerick 1 and 2 Proposed Technical Specification Changes Common to Reactor Protection System or ECCS Actuation Instrumentation".

b) Enclosure 2 - GE Document No. MFN 024-90, OG90-319-32D, letter from W. P. Sullivan, GE, to Millard L. Wohl, NRC, dated March 22, 1990: "Clarification of Technical Specification Changes Given in ECCS Actuation Instrumentation Analysis)".

c) Enclosure 3 - GE Document No. MDE 83-0485-3, "Technical Specification Improvement Analysis for the Reactor Protection System for LaSalle County Station Units 1 and 2", dated April, 1991.

d) Enclosure 4 - GE document No. RE-025-1, "Technical Specification Improvement Analysis for Emergency Core Cooling System Actuation Instrumentation for LaSalle County Station Units 1 and 2", dated April, 1991.

DRAFT

DRAFT

Please note that applications and accompanying affidavits in accordance with 10 CFR 2.790(b)(1), to withhold from public disclosure GE Document Nos. MDE-83-0485-3, "Technical Specification Improvement Analysis for the Reactor Protection System for LaSalle County Station Units 1 and 2", dated April, 1991, and RE-025-1, "Technical Specification Improvement Analysis for Emergency Core Cooling System Actuation Instrumentation for LaSalle County Station Units 1 and 2", dated April, 1991, are included with Enclosures 3 and 4, respectively. Accordingly, we request that the documents identified above (Enclosures 3 and 4) be withheld from public disclosure in accordance with 10 CFR 2.790(a)(4).

If you have any questions regarding this matter, please contact us.

Sincerely yours,

Attachments

Enclosures

DRAFT

ATTACHMENT 1

LASALLE COUNTY STATION
UNITS 1 AND 2

DOCKET NOS. ______________

LICENSE NOS. ______________

TECHNICAL SPECIFICATIONS CHANGE REQUEST

"REDUCED TESTING OF REACTOR PROTECTION SYSTEM, EMERGENCY CORE COOLING SYSTEM, ISOLATION ACTUATION AND COMMON INSTRUMENTATION"

Supporting Information for Changes - 6 pages

Commonwealth Edison Company (CECO), Licensee under Facility Operating Licenses ___________ and ___________ for LaSalle County Station (LSCS), Unit 1 and Unit 2, respectively, hereby requests that the Technical Specifications (TS) contained in Appendix A of the Operating Licenses be amended as proposed herein to extend surveillance test intervals (STIs) and allowable out-of-service times (AOTs) for the actuation instrumentation supporting Reactor Protections System (RPS), Emergency Core Cooling System (ECCS), and Isolation Actuation, including instrumentation common to the Control Rod Block Function (CRBF), the Reactor Core Isolation Cooling (RCIC) system, End-of-Cycle Recirculation Pump Trip (EOC-RPT) system, and the isolation instrumentation common to RPS and/or ECCS. The proposed changes will minimize unnecessary testing and remove excessive restrictive AOTs that could potentially degrade overall plant safety and availability.

We request the changes proposed herein to be effective fifteen (15) days after issuance of the Amendments.

This Change Request provides a discussion and description of the proposed TS changes and a safety assessment of the proposed TS changes.

Discussion and Description of Proposed Changes

Licensing Topical Report (LTR), "BWR Owners' Group Response to NRC Generic Letter 83-28, Item 4.5.3", (Reference 1) provided justification for the acceptability of current RPS STIs. In addition, Reference 1 established a basis for extending STIs and AOTs for RPS based on reliability analyses which estimate RPS failure frequency. The analyses were further developed in other LTRs (References 2 through 6) to provide justification for extending TS STIs and AOTs for the RPS, ECCS, and Isolation Actuation including common instrumentation. References 2 through 6 also included proposed TS changes to facilitate implementation of the analyses results. References 2 through 6 were submitted to the NRC by the Boiling Water Reactor Owners' Group (BWROG) and subsequently approved as detailed in NRC Safety Evaluation Reports (SERs). These SERs describe the acceptability of both the analyses and the proposed TS changes provided to the NRC. In addition, the NRC SERs provided criteria for plant specific implementation of the generically approved TS changes. Our compliance with these criteria is discussed in the Safety Assessment of this Change Request.

The Change Request proposed TS changes to the actuation instrumentation supporting the RPS, ECCS, and Isolation Actuation including instrumentation common to the CRBF and the isolation instrumentation common to the RPS and/or ECCS. These changes are specifically designated in the TS mark-ups of References 2 through 6 and therefore, are not further discussed here. We are also proposing TS changes to instrumentation common to RPS and/or ECCS, but which are not specifically designated in References 2 through 6. These proposed changes are addressed in the analyses of References 2 through 6, but were not specifically designated in the TS mark-ups submitted as part of References 2 through 6. These changes will provide a complete consideration of all systems/components initiated by RPS, ECCS, or Isolation Actuation instrumentation which are tested on a monthly schedule and are NRC approved for testing on a quarterly schedule as detailed in References 2 through 6. All changes are shown in Attachment 3. However, only those changes not specifically designated in References 2 through 6 are described below.

1. The EOC-RPT system uses trip functions common to RPS. Therefore, we propose to change the EOC-RPT system STIs and AOTs on TS pages 3/4 3-39, 3/4 3-41, and 3/4 3-44 to conform to the TS changes made for RPS instrumentation. Enclosure 1 details the fact that the analysis of Reference 2 bounds the proposed TS changes to EOC-RPT.

2. The RCIC system uses trip functions common to ECCS and therefore, we propose changes to TS pages 3/4 3-46, 3/4 3/47, 3/4 3-49 to be consistent with other TS changes for ECCS instrumentation. Changes to these TS were not specifically included in the TS mark-ups provided to the NRC in Reference 5, although they are addressed in the Reference 5 analysis. Enclosure 2, GE document No. MFN 024-90, letter from W. P. Sullivan, GE, to Millard L. Wohl, NRC, dated March 22, 1990, "Clarification of Technical Specification Changes Given in ECCS Actuation Instrumentation Analysis", provided mark-ups for these changes to the RCIC system TS to the BWROG TSC.

3. The proposed TS changes to TS Section 3/4 3.3.3 on page 3/4 3-27 and 27(a) provide a 24 hour AOT for ECCS instrumentation which is consistent with the analysis in Reference 5. The proposed wording differs from the TS mark-up of Reference 5 which implies an allowance of 24 hours before taking the action of TS Table 3.3.3-1. Enclosure 2 provides a clarification on the intent of the Reference 5 TS mark-up and also provides revised wording. We have proposed a TS change consistent with Enclosure 2.

Safety Assessment

The effect on safety of the proposed extensions to the STIs and AOTs of the actuation instrumentation supporting the RPS, ECCS, and Isolation Actuation including the instrumentation common to the CRBF, and the isolation instrumentation common to the RPS and/or ECCS has been addressed in References 2 through 6. Further, the NRC has detailed their acceptance of the analyses and conclusions of References 2 through 6 in SERs (included in References 2 through 6). The SERs conclude that implementation of the TS changes proposed in References 2 through 6 would provide an overall enhancement to plant safety and that the proposed changes to TS are acceptable subject to the Licensee documenting 1) plant-specific applicability, 2) that instrument drift is bounded by the generic analysis assumptions, and 3) confirmation that differences between plant specific and generic RPSs were included in the plant-specific analysis. These acceptance conditions are addressed below.

1. A plant-specific review of the LTR's (References 2 through 6) applicability to LSCS has been conducted. For the RPS, the review

compared the LSCS RPS configuration and surveillance test procedure with the generic RPS evaluated in the LTR. The differences between the two were identified and the reliability effect of the differences was assessed. The differences and their effect are documented in a separate GE report, Enclosure 3, document 83-0485-3, "Technical Specification Improvement Analysis for the Reactor Protection System for LaSalle County Station Units 1 and 2", dated April, 1991. The report identifies differences which were dispositioned by either engineering assessment or additional analyses. The report concluded that these differences would not significantly affect the improvement in plant safety which would be obtained through the TS changes evaluated in the generic analysis and that the generic analysis is applicable to LSCS Units 1 and 2.

For ECCS, a similar review was conducted. The results are documented in a separate GE report, Enclosure 4, document No. RE-025-1, "Technical Specification Improvement Analysis for Emergency Core Cooling System Actuation Instrumentation for LaSalle County Station Units 1 and 2," dated April, 1991. The report concludes that the ECCS configuration for LSCS is similar to the generic analysis with only seven differences. The differences were dispositioned by engineering assessment or additional analyses. The results indicate that the proposed changes to ECCS instrumentation would meet the acceptance criterion of Reference 5, Part 2. The changes to the instrumentation common to the CRBF and the isolation instrumentation common to the RPS and/or ECCS are addressed in References 3 and 4, respectively. These changes are bounded by the generic analyses in References 2 and 5 and LSCS plant-specific analyses given in Enclosures 1 and 3.

For Isolation Actuation instrumentation, analyses were provided in Reference 6 that bounded the plant-specific differences. Appendix C of Reference 6 provides the LSCS Unit 1 and 2 STIs and calibration intervals that were included in the study.

2. In 1988, the NRC issued additional guidance regarding instrument drift (Reference 7). This letter states that "licensees need only confirm that the setpoint drift which could be expected under the extended STIs has been studied and either (1) has been shown to remain within the existing allowance in the RPS (for BWRs) instrument setpoint calculation or (2) that the allowance and setpoint have been adjusted to account for the additional expected drift". Present setpoint calculations for LSCS are based on an eighteen (18) month calibration interval. Therefore, drift occurring during a three-month STI falls within the existing drift allowance.

3. We have reviewed the GE plant-specific report for LSCS and have verified that the differences between the LSCS and generic RPS were included in the plant-specific analysis. Therefore, the generic analysis in Reference 2 is applicable to LSCS.

As discussed above, we have conformed to the guidance provided in References 2 through 6 in the three areas to be addressed by Licensees to ensure the acceptability of proposed TS changes. As noted previously, several changes are also proposed which are not specifically referenced in the NRC SERs given in References 2 through 6. The following discussion addresses the acceptability of these proposed changes.

1. The EOC-RPT is initiated by signals common to the RPS. These signals (turbine stop valve closure and turbine control valve low hydraulic pressure) were not identified as common trip functions in the RPS TS improvement analysis (Reference 2). Although STI changes to the common EOC-RPT trip functions were not explicitly identified in the Reference 2 analysis, the changes can be considered bounded by this analysis. The basis for this conclusion is similar to the basis established in Reference 3.

2. Analysis of the effects of extending AOTs and STIs for the RCIC system instrumentation was completed and found acceptable as detailed in Reference 5. However, proposed changes to the TS were not specifically given in Reference 5. This does not affect the acceptability of these proposed changes, since the methods and results of Reference 5 were found acceptable as documented in Reference 5.

   Recognizing that mark-ups to the RCIC system instrumentation TS had not been previously included in Reference 5, GE provided TS mark-ups for all GE BWR product lines, incorporating the extended STIs and AOTs for RCIC system instrumentation. The mark-ups were provided in Enclosure 2.

3. Also discussed in Enclosure 2 is a clarification of the applicability of the 24-hour TS AOT for ECCS Actuation Instrumentation. The change provides a 24-hour AOT in those TS Action Statements which are applicable to specific instrumentation. The intent of the change is to preclude the allowance of 24 hours before taking the action specified in TS Table 3.3.3.1. Action "b" of TS paragraph 3/4 3.3, as written in Reference 5 implies a 24 hour AOT before taking any action in TS Table 3.3.3-1. The change we have proposed accurately reflects the intent of the Reference 5 analysis. This change, therefore, is necessary to obtain the overall enhancement to safety that is possible by extending STIs and AOTs.

   References 2 through 6 provided TS changes based on review of the LTRs. We have proposed TS changes consistent with those previously approved and specifically designated in References 2 through 6. In addition, several changes are proposed which are not explicitly referenced in the NRC SERs, but are covered by the analyses detailed in References 2 through 6 and Enclosures 3 and 4, and are acceptable as discussed above.

   In summary, the NRC criteria for demonstrating the applicability and acceptability of all proposed changes has been shown to be met, as detailed above. We, therefore, conclude that the changes proposed will minimize unnecessary testing and relax excessively restrictive AOTs, and will provide an overall enhancement to plant safety.

<u>Information Supporting a Finding of No Significant Hazards Consideration</u>

We have concluded that the proposed changes to the LSCS TS, which extend STIs and AOTs for the RPS, ECCS, and Isolation Actuation instrumentation including instrumentation common to RPS and/or ECCS, do not constitute a Significant Hazards Consideration. In support of this determination, an evaluation of each of the three standards set forth in 10 CFR 50.92 is provided below.

1) The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed TS changes increase the STIs and AOTs for actuation instrumentation supporting RPS, ECCS, and Isolation Actuation including instrumentation common to the CRBF, RCIC system, EOC-RPT, and isolation functions. There are no changes in any of the affected systems themselves. Since there are no such changes, there can be no change in the probability of occurrence of an accident or the consequences of an accident or the consequences of malfunction of equipment. Regarding the probability of malfunction of equipment, LTRs prepared by GE showed that for the RPS there is a reduction in scram frequency, but that in the ECCS case, there is a small increase in the unavailability of the water injection function. This increase in unavailability was judged acceptable by GE. The NRC in its review of the LTRs (References 2 through 6), concurred with this conclusion. The changes proposed are consistent with these SERs given in References 2 through 6 with several additions. These additional changes are bounded by the analyses of References 2 through 6 as detailed in this Change Request and in Enclosures 1 and 2. Therefore, the proposed changes do not involve a significant increase in the probability of consequences of an accident previously evaluated.

2) The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed TS changes do not create the possibility for an accident or malfunction of a different type than any evaluated previously in the Final Safety Analysis Report (FSAR). The proposed changes increase the STIs and AOTs for the RPS, ECCS, and Isolation Actuation instrumentation including common instrumentation. There are no changes in the RPS, ECCS, Isolation Actuation or common systems themselves. Since there are no such changes, there is no possibility for an accident or malfunction of a different type than any evaluated previously.

3) The proposed changes do not involve a significant reduction in a margin of safety.

   The proposed TS changes do not reduce the margin of safety as defined in the basis for any TS. The proposed TS changes do not change any setpoints in the RPS, ECCS, Isolation Actuation instrumentation, or common systems, or their levels of redundancy. Setpoints are based upon the drift occurring during the 18-month calibration interval. The proposed changes extend STIs and AOTs. The bases in the TS either do not discuss STIs, or state "...one channel may be inoperable for brief intervals to conduct required surveillance." The proposed TS changes discussed in References 2 through 6 as well as the additional changes discussed in this Change Request and Enclosures 1 and 2, are bounded by the analyses in References 2 through 6. These analyses (References 2 through 6) prepared by GE and reviewed and approved by the NRC examined the effects of extending STIs and AOTs and found that the proposed changes would not involve a significant reduction in a margin of safety.

Information Supporting an Environment Assessment

An environmental assessment is not required for the changes proposed by the Change Request because the requested changes conform to the criteria for "actions eligible for categorical exclusion", as specified in 10CFR 51.22(c)(9). The requested changes will have no impact on the environment. The proposed changes do not involve a significant hazards consideration as discussed in the preceding section. The proposed changes do not involve a significant change in the types or significant increase in the amounts of any effluents that may be released offsite. In addition, the proposed changes do not involve a significant increase in individual or cumulative occupational radiation exposure.

Conclusion

The (To Be Provided by CECO) have reviewed these proposed changes to the TS and determined that they do not involve an Unreviewed Safety Question and will not endanger the health and safety of the public.

ATTACHMENT 2

REFERENCES

1. S. Visweswaran, et al., "BWR Owners' Group Response to NRC Generic Letter 83-28, Item 4.5.3, "General Electric Company, NEDC-30844A, March 1988

2. W. P. Sullivan, et al., "Technical Specification Improvement Analyses for BWR Reactor Protection System", General Electric Company, NEDC-30851P-A, March, 1988

3. S. Visweswaran, et al., "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation", General Electric Company, NEDC-30851P-A, Supplement 1, October, 1988

4. L. G. Frederick, et al., "Technical Specification Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation", General Electric Company, NEDC-30851P-A, Supplement 2, March, 1989

5. D. B. Atcheson, et al., "BWR Owners Group Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation", Parts 1 and 2, General Electric Company, NEDC-30936P-A, December, 1988

6. W. P. Sullivan, et al., "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation", General Electric Company, NEDC-31677OP-A, July, 1990

7. C. E. Rossi, NRC, to R. F. Janacek, BWROG, "Staff Guidance for Licensee Determination that the Drift Characteristics for Instrumentation Used in RPS Channels are Bounded by NEDC-30851P Assumptions when the Functional Test Interval is Extended from Monthly to Quarterly", April 27, 1988

ATTACHMENT 3

PROPOSED TECHNICAL SPECIFICATION CHANGES
FOR LASALLE COUNTY STATION

List of Attached Change Pages for LaSalle County Station Unit 1 Technical Specifications

3/4 3-1
3/4 3-5
3/4 3-7
3/4 3-8
3/4 3-9
3/4 3-14
3/4 3-20
3/4 3-21
3/4 3-22
3/4 3-26
3/4 3-27
3/4 3-27(a)
3/4 3-32
3/4 3-33
3/4 3-34
3/4 3-39
3/4 3-41
3/4 3-44
3/4 3-46
3/4 3-47
3/4 3-49
3/4 3-50
3/4 3-52
3/4 3-54

B 3/4 3-1
B 3/4 3-2
B 3/4 3-3
B 3/4 3-4

The markups are based on Unit 1 Technical Specifications, Amendment 75 plus Modifications #M-1-1-87-096 and #M-1-1-87-097.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION: One required channel inoperable for one or more Functional Units,

a. With ~~the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system,~~ place the inoperable channels and/or that trip system in the tripped condition* within ~~1 hour.~~ 12 hours, otherwise take the ACTION required by Table 3.3.1-1 ~~The provisions of Specification 3.0.4 are not applicable.~~

Insert (A) →

~~b. With the the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within 1 hour and take the ACTION required by Table 3.3.1-1.~~

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1 1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.~~***~~ **

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip functional unit shown in Table 3.3.1-2 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip system.

Insert (B) →

~~*With a design providing only one channel per trip system, an inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.1-1 for that Trip Function shall be taken.~~

~~**If more channels are inoperable in one trip system than in the other, select that trip system to place in the tripped condition, except when this would cause the Trip Function to occur.~~

** ~~***~~The specified 18-month interval may be waived for Cycle 1 provided the surveillance is performed during Refuel 1, which is to commence no later than October 27, 1985.

Insert "A" to LSCS Technical Specification 3.3.1.b

b. With the number of OPERABLE channels less than allowed by Action a.;

1. ensure each required Functional Unit maintains trip capability in each Trip System within 1 hour,

2. ensure for each required Functional Unit, the minimum OPERABLE Channels per Trip system in one trip system and/or the trip system are OPERABLE or in the tripped condition within 6 hours,

3. place all inoperable channels and/or associated trip system(s) in the tripped condition* within 12 hours,

otherwise take the ACTION required by Table 3.3.1-1.

Insert "B" to LSCS Technical Specification 3.3.1.b

* An inoperable channel and/or the trip system need not be placed in the tripped condition if this would cause the Trip Function to occur. When a trip system can be placed in the tripped condition without causing the Trip Function to occur, place the most degraded trip system in the tripped condition; if both systems have the same number of inoperable channels, place either trip system in the tripped condition.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

6

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the channel in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(b) The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn* and during shutdown margin demonstrations performed per Specification 3.10.3.

(c) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than 14 LPRM inputs to an APRM channel.

(d) This function is not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.10.1.

(e) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(f) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(g) Also actuates the standby gas treatment system.

(h) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(i) This function shall be automatically bypassed when turbine first stage pressure is $\leq$ 140 psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER.

(j) Also actuates the EOC-RPT system.

*Not required for control rods removed per Specifications 3.9.10.1 or 3.9.10.2.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors				
a. Neutron Flux - High	S/U[b], S	S/U[c], W	R	2
	S	W	R	3, 4, 5
b. Inoperative	NA	W	NA	2, 3, 4, 5
2. Average Power Range Monitor:[f]				
a. Neutron Flux - High, Setdown	S/U[b], S	S/U[c], W	SA	1, 2
	S	W	SA	3, 5
b. Flow Biased Simulated Thermal Power-Upscale	S, D[g]	S/U[c], ~~W~~ Q	W[d][e], SA, R[h]	1
c. Fixed Neutron Flux - High	S	S/U[c], ~~W~~ Q	W[d], SA	1
d. Inoperative	NA	~~W~~ Q	NA	1, 2, 3, 5
3. Reactor Vessel Steam Dome Pressure - High	NA	~~M~~ Q	Q	1, 2
4. Reactor Vessel Water Level - Low, Level 3	~~NA~~ S	~~M~~ Q	R	1, 2
5. Main Steam Line Isolation Valve - Closure	NA	~~M~~ Q	R	1
6. Main Steam Line Radiation - High	S	~~M~~ Q	R	1, 2
7. Primary Containment Pressure - High	NA	~~M~~ Q	Q	1, 2

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High	NA	~~M~~ → Q	R	1, 2, 5
9. Turbine Stop Valve - Closure	NA	~~M~~ → Q	R	1
10. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	NA	~~M~~ → Q	R*	1
11. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	~~M~~ → W	NA	1, 2, 3, 4, 5
13. Control Rod Drive				
a. Charging Water Header Pressure - Low	NA	M	R	2, 5
b. Delay Timer	NA	M	R	2, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) The IRM, and SRM channels shall be determined to overlap for at least 1/2 decades during each startup and the IRM and APRM channels shall be determined to overlap for at least 1/2 decades during each controlled shutdown, if not performed within the previous 7 days.

(c) Within 24 hours prior to startup, if not performed within the previous 7 days.

(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power levels calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER $\geq$ 25% of RATED THERMAL POWER. The APRM Gain Adjustment Factor (GAF) for any channel shall be equal to the power value determined by the heat balance divided by the APRM reading for that channel.

Within 2 hours, adjust any APRM channel with a GAF > 1.02. In addition, adjust any APRM channel within 12 hours, (1) if power is greater than or equal to 90% of RATED THERMAL POWER and the APRM channel GAF is < 0.98, or (2) if power is less than 90% of RATED THERMAL POWER and the APRM reading exceeds the power value determined by the heat balance by more than 10% of RATED THERMAL POWER. Until any required APRM adjustment has been accomplished, notification shall be posted on the reactor control panel.

(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH).

(g) Measure and compare core flow to rated core flow.

(h) This calibration shall consist of verifying the 6 $\pm$ 1 second simulated thermal power time constant.

*The specified 18-month interval may be waived for Cycle 1 provided the surveillance is performed during Refuel 1, which is to commence no later than October 27, 1985.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The isolation actuation instrumentation channels shown in Table 3.3.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.2-2 and with ISOLATION SYSTEM RESPONSE TIME as shown in Table 3.3.2-3.

APPLICABILITY: As shown in Table 3.3.2-1.

ACTION:

a. With an isolation actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

Insert (A) →

~~b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the operable channel(s) and/or trip system in the tripped condition* within one hour. The provisions of Specification 3.0.4 are not applicable.~~

*

c. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition*** within one hour and take the ACTION required by Table 3.3.2-1.

Insert (B) →

~~*An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by by Table 3.3.2-1 for that Trip Function shall be taken.~~

~~**If more channels are inoperable in one trip system than in the other, select that trip system to place in the tripped condition except when this would cause the Trip Function to occur.~~

~~***An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 1 hour or the ACTION required by Table 3.3.2-1 for that Trip Function shall be taken.~~

Insert "A" to LSCS Technical Specification 3.3.2.b

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system:

   1. If placing the inoperable channel(s) in the tripped condition would cause an isolation, the inoperable channel(s) shall be restored to OPERABLE status within 6 hours

      or take the ACTION required by Table 3.3.2-1 for the affected trip function shall be taken.

   2. If placing the inoperable channel(s) in the tripped condition would not cause an isolation, the inoperable channel(s) and/or that trip system shall be placed in the tripped condition within: a) 12 hours for trip functions common to RPS Instrumentation; and b) 24 hours for trip functions not common to RPS Instrumentation.

The provisions of Specification 3.0.4 are not applicable.

Insert "B" to LSCS Technical Specification 3.3.2.c

*

Place one trip system (with the most inoperable channels) in the tripped condition. The trip system need not be placed in the tripped condition when this would cause the isolation to occur.

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

ACTION

ACTION 20 - Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN with the next 24 hours.

ACTION 21 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 22 - Close the affected system isolation valves within 1 hour and declare the affected system inoperable.

ACTION 23 - Be in at least STARTUP within 6 hours.

ACTION 24 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within 1 hour.

ACTION 25 - Lock the affected system isolation valves closed within 1 hour and declare the affected system inoperable.

ACTION 26 - Provided that the manual initiation function is OPERABLE for each other group valve, inboard or outboard, as applicable, in each line, restore the manual initiation function to OPERABLE status within 24 hours; otherwise, restore the manual initiation function to OPERABLE status within 8 hours; otherwise:

a. Be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, or

b. Close the affected system isolation valves within the next hour and declare the affected system in operable.

NOTES

* May be bypassed with reactor steam pressure ≤ 1043 psig and all turbine stop valves closed.

** When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

\# During CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

(a) See Specification 3.6.3, Table 3.6.3-1 for valves in each valve group.

(b) A channel may be placed in an inoperable status for up to 2 6 hours for required surveillance without placing the channel in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter. In addition for those trip systems with a design providing only one channel per trip system, the channel may be placed in an inoperable status for up to 8 hours for required surveillance testing without placing the channel in the tripped condition provided that the redundant isolation valve, inboard or outboard, as applicable, in each line is operable and all required actuation instrumentation for that redundant valve is OPERABLE, or place the trip system in the tripped condition.

(c) Also actuates the standby gas treatment system.

(d) A channel is OPERABLE if 2 of 4 instruments in that channel are OPERABLE.

(e) Also actuates secondary containment ventilation isolation dampers per Table 3.6.5.2-1.

(f) Closes only RWCU system inlet outboard valve.

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
A. AUTOMATIC INITIATION				
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level				
1) Low, Level 3	NA	~~M~~ Q	R	1, 2, 3
2) Low Low, Level 2	NA	~~M~~ Q	R	1, 2, 3
3) Low Low Low, Level 1	NA	~~M~~ Q	R	1, 2, 3
b. Drywell Pressure - High	NA	~~M~~ Q	Q	1, 2, 3
c. Main Steam Line				
1) Radiation - High	S	~~M~~ Q	R	1, 2, 3
2) Pressure - Low	NA	~~M~~ Q	Q	1
3) Flow - High	NA	~~M~~ Q	R	1, 2, 3
d. Main Steam Line Tunnel Temperature - High	NA	~~M~~ Q	R	1, 2, 3
e. Condenser Vacuum - Low	NA	~~M~~ Q	Q	1, 2*, 3*
f. Main Steam Line Tunnel Δ Temperature - High	NA	~~M~~ Q	R	1, 2, 3
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Building Vent Exhaust Plenum Radiation - High	S	~~M~~ Q	R	1, 2, 3 and **
b. Drywell Pressure - High	NA	~~M~~ Q	Q	1, 2, 3
c. Reactor Vessel Water Level - Low Low, Level 2	NA	~~M~~ Q	R	1, 2, 3, and #
d. Fuel Pool Vent Exhaust Radiation - High	S	~~M~~ Q	R	1, 2, 3 and **
3. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	S	~~M~~ Q	R	1, 2, 3
b. Heat Exchanger Area Temperature - High	NA	~~M~~ Q	Q	1, 2, 3
c. Heat Exchanger Area Ventilation ΔT - High	NA	~~M~~ Q	Q	1, 2, 3
d. SLCS Initiation	NA	R	NA	1, 2, 3
e. Reactor Vessel Water Level - Low Low, Level 2	NA	~~M~~ Q	R	1, 2, 3

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
4. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow - High	NA	~~M~~→Q	Q	1, 2, 3
b. RCIC Steam Supply Pressure - Low	NA	~~M~~→Q	Q	1, 2, 3
c. RCIC Turbine Exhaust Diaphragm Pressure - High	NA	~~M~~→Q	Q	1, 2, 3
d. RCIC Equipment Room Temperature - High	NA	~~M~~→Q	Q	1, 2, 3
e. RCIC Steam Line Tunnel Temperature - High	NA	~~M~~→Q	Q	1, 2, 3
f. RCIC Steam Line Tunnel Δ Temperature - High	NA	~~M~~→Q	Q	1, 2, 3
g. Drywell Pressure - High	NA	~~M~~→Q	Q	1, 2, 3
h. RCIC Equipment Room Δ Temperature - High	NA	~~M~~→Q	Q	1, 2, 3
5. RHR SYSTEM STEAM CONDENSING MODE ISOLATION				
a. RHR Equipment Area Δ Temperature - High	NA	~~M~~→Q	Q	1, 2, 3
b. RHR Area Cooler Temperature - High	NA	~~M~~→Q	Q	1, 2, 3
c. RHR Heat Exchanger Steam Supply Flow - High	NA	~~M~~→Q	Q	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
6. RHR SYSTEM SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 3	NA	~~M~~ → Q	R	1, 2, 3
b. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA	~~M~~ → Q	Q	1, 2, 3
c. RHR Pump Suction Flow - High	NA	~~M~~ → Q	Q	1, 2, 3
d. RHR Area Temperature - High	NA	~~M~~ → Q	Q	1, 2, 3
e. RHR Equipment Area ΔT - High	NA	~~M~~ → Q	Q	1, 2, 3
B. MANUAL INITIATION				
1. Inboard Valves	NA	R	NA	1, 2, 3
2. Outboard Valves	NA	R	NA	1, 2, 3
3. Inboard Valves	NA	R	NA	1, 2, 3 and **,#
4. Outboard Valves	NA	R	NA	1, 2, 3 and **,#
5. Inboard Valves	NA	R	NA	1, 2, 3
6. Outboard Valves	NA	R	NA	1, 2, 3
7. Outboard Valve	NR	R	NA	1, 2, 3

*When reactor steam pressure > 1043 psig and/or any turbine stop valve is open.

**When handling irradiated fuel in the secondary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

#During CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION(a)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
C. DIVISION 3 TRIP SYSTEM			
1. HPCS SYSTEM			
a. Reactor Vessel Water Level - Low, Low, Level 2	4(b)	1, 2, 3, 4*, 5*	35
b. Drywell Pressure - High	4(b)	1, 2, 3	35
c. Reactor Vessel Water Level-High, Level 8	2(c)	1, 2, 3, 4*, 5*	32
d. Condensate Storage Tank Level-Low	2(d)	1, 2, 3, 4*, 5*	36
e. Suppression Pool Water Level-High	2(d)	1, 2, 3, 4*, 5*	36
f. Pump Discharge Pressure-High (Bypass)	1	1, 2, 3, 4*, 5*	31
g. HPCS System Flow Rate-Low (Permissive)	1	1, 2, 3, 4*, 5*	31
h. Manual Initiation	1/division	1, 2, 3, 4*, 5*	34

D. LOSS OF POWER

	TOTAL NO. OF INSTRUMENTS	INSTRUMENTS TO TRIP	MINIMUM OPERABLE INSTRUMENTS(a)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	2/bus	2/bus	2/bus	1, 2, 3, 4**, 5**	37
2. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	2/bus	2/bus	2/bus	1, 2, 3, 4**, 5**	37

6

(a) A channel instrument may be placed in an inoperable status for up to 2 hours during periods of required surveillance without placing the trip system/channel/instrument in the tripped condition provided at least one other OPERABLE channel/instrument in the same trip system is monitoring that parameter.
(b) Also actuates the associated division diesel generator.
(c) Provides signal to close HPCS pump discharge valve only on 2-out-of-2 logic.
(d) Provides signal to HPCS pump suction valves only.
* Applicable when the system is required to be OPERABLE per Specification 3.5.2 or 3.5.3.
** Required when ESF equipment is required to be OPERABLE.
\# Not required to be OPERABLE when reactor steam dome pressure is $\leq$ 122 psig.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION

ACTION 30 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:

a. With one channel inoperable, place the inoperable channel in the tripped condition within one hour* or declare the associated system inoperable. 24 hours*

b. With more than one channel inoperable, declare the associated system inoperable.

24 hours

ACTION 31 - With the number of OPERABLE channels less than required by the Minimum OPERABLE channels per Trip Function, place the inoperable channel in the tripped condition within one hour, restore the inoperable channel to OPERABLE status within 7 days or declare the associated system inoperable.

ACTION 32 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, declare the associated ADS trip system or ECCS inoperable. within 24 hours.

ACTION 33 - With the number of OPERABLE channels less than the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within ~~one hour.~~ 24 hours.

ACTION 34 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 8 hours or declare the associated ADS trip system or ECCS inoperable. 24

ACTION 35 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement

a. For one trip system, place that trip system in the tripped condition within one hour* or declare the HPCS system inoperable. 24 hours*

b. For both trip systems, declare the HPCS system inoperable.

ACTION 36 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour* or declare the HPCS system inoperable. 24 hours*

ACTION 37 - With the number of OPERABLE instruments less than the Minimum Operable Instruments, place the inoperable instrument(s) in the tripped condition within 1 hour* or declare the associated emergency diesel generator inoperable and take the ACTION required by Specification 3.8.1.1 or 3.8.1.2 as appropriate.

*The provisions of Specification 3.0.4 are not applicable.

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION

ACTION 38 With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per trip function requirements:

a. With one channel inoperable, remove the inoperable channel within one hour, restore the inoperable channel to OPERABLE status within 7 days or declare the associated ECCS systems inoperable. 24 hours

b. With both channels inoperable, restore at least one channel to OPERABLE status within one hour or declare the associated ECCS systems inoperable.

TABLE 4.3.3.1-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
A. DIVISION I TRIP SYSTEM				
1. RHR-A (LPCI MODE) AND LPCS SYSTEM				
a. Reactor Vessel Water Level - Low Low Low, Level 1	~~NA~~ S	~~M~~ Q	R	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	NA	~~M~~ Q	Q	1, 2, 3
c. LPCS Pump Discharge Flow-Low	NA	~~M~~ Q	Q	1, 2, 3, 4*, 5*
d. LPCS and LPCI A Injection Valve Injection Line Pressure Low Interlock	NA	~~M~~ Q	R	1, 2, 3, 4*, 5*
e. LPCS and LPCI A Injection Valve Reactor Pressure Low Interlock	NA	~~M~~ Q	R	1, 2, 3, 4*, 5*
f. LPCI Pump A Start Time Delay Relay	NA	~~M~~ Q	Q	1, 2, 3, 4*, 5*
g. LPCI Pump A Flow-Low	NA	~~M~~ Q	Q	1, 2, 3, 4*, 5*
h. Manual Initiation	NA	R***	NA	1, 2, 3, 4*, 5*
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"#				
a. Reactor Vessel Water Level - Low Low Low, Level 1	~~NA~~ S	~~M~~ Q	R	1, 2, 3
b. Drywell Pressure-High	NA	~~M~~ Q	Q	1, 2, 3
c. Initiation Timer	NA	~~M~~ Q	Q	1, 2, 3
d. Reactor Vessel Water Level - Low, Level 3	~~NA~~ S	~~M~~ Q	R	1, 2, 3
e. LPCS Pump Discharge PRessure-High	NA	~~M~~ Q	Q	1, 2, 3
f. LPCI Pump A Discharge Pressure-High	NA	~~M~~ Q	Q	1, 2, 3
g. Manual Initiation	NA	R	NA	1, 2, 3
h. Drywell Pressure Bypass Timer	NA	~~M~~ Q	Q	1, 2, 3
i. Manual Inhibit	NA	R	NA	1, 2, 3

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
B. DIVISION 2 TRIP SYSTEM				
1. RHR B AND C (LPCI MODE)				
a. Reactor Vessel Water Level - Low Low Low, Level 1	~~NA~~ S	~~M~~→ Q	R	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	NA	~~M~~→ Q	Q	1, 2, 3
c. LPCI B and C Injection Valve Injection Line Pressure Low Interlock	NA	~~M~~→ Q	R	1, 2, 3, 4*, 5*
d. LPCI Pump B Start Time Delay Relay	NA	~~M~~→ Q	Q	1, 2, 3, 4*, 5*
e. LPCI Pump Discharge Flow-Low	NA	~~M~~→ Q	Q	1, 2, 3, 4*, 5*
f. Manual Initiation	NA	R***	NA	1, 2, 3, 4*, 5*
g. LPCI B and C Injection Valve Reactor Pressure Low Interlock	NA	~~M~~→ Q	R	1, 2, 3, 4*, 5*
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"#				
a. Reactor Vessel Water Level - Low Low Low, Level 1	~~NA~~ S	~~M~~→ Q	R	1, 2, 3
b. Drywell Pressure-High	NA	~~M~~→ Q	Q	1, 2, 3
c. Initiation Timer	NA	~~M~~→ Q	Q	1, 2, 3
d. Reactor Vessel Water Level - Low, Level 3	~~NA~~ S	~~M~~→ Q	R	1, 2, 3
e. LPCS Pump B and C Discharge Pressure-High	NA	~~M~~→ Q	Q	1, 2, 3
f. Manual Initiation	NA	R	NA	1, 2, 3
h. Drywell Pressure Bypass Timer	NA	~~M~~→ Q	Q	1, 2, 3
i. Manual Inhibit	NA	R	NA	1, 2, 3

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
C. DIVISION 3 TRIP SYSTEM				
1. HPCS SYSTEM				
a. Reactor Vessel Water Level - Low Low, Level 2	~~NA~~ S	~~M~~ → Q	R	1, 2, 3, 4*, 5*
b. Drywell Pressure-High	NA	~~M~~ → Q	Q	1, 2, 3
c. Reactor Vessel Water Level-High Level 8	~~NA~~ S	~~M~~ → Q	R	1, 2, 3, 4*, 5*
d. Condensate Storage Tank Level - Low	NA	~~M~~ → Q	Q	1, 2, 3, 4*, 5*
e. Suppression Pool Water Level - High	NA	~~M~~ → Q	Q	1, 2, 3, 4*, 5*
f. Pump Discharge Pressure-High	NA	~~M~~ → Q	Q	1, 2, 3, 4*, 5*
g. HPCS System Flow Rate-Low	NA	~~M~~ → Q	Q	1, 2, 3, 4*, 5*
h. Manual Initiation	NA	R	NA	1, 2, 3, 4*, 5*
D. LOSS OF POWER				
1. 4.16 kV Emergency Bus Under-voltage (Loss of Voltage)	NA	NA	R	1, 2, 3, 4**, 5**
2. 4.16 kV Emergency Bus Under-voltage (Degraded Voltage)	NA	NA	R	1, 2, 3, 4**, 5**

#Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 122 psig.

*When the system is required to be OPERABLE after being manually realigned, as applicable, per Specification 3.5.2.

**Required when ESF equipment is required to be OPERABLE.

***The specified 18-month interval may be waived for Cycle 1 provided the surveillance is performed during Refuel 1, which is to commence no later than October 27, 1985.

INSTRUMENTATION

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.2 The end-of-cycle recirculation pump trip (EOC-RPT) system instrumentation channels shown in Table 3.3.4.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4.2-2 and with the END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME as shown in Table 3.3.4.2-3.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 30% of RATED THERMAL POWER.

ACTION:

a. With an end-of-cycle recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within 1 hour. 12 hours.

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement(s) for one trip system and:

    1. If the inoperable channels consist of one turbine control valve channel and one turbine stop valve channel, place both inoperable channels in the tripped condition within 1 hour. 12 hours.

    2. If the inoperable channels include two turbine control valve channels or two turbine stop valve channels, declare the trip system inoperable.

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours. Otherwise, either:

    1. Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Limiting Condition for Operation (LCO) to the EOC-RPT inoperable value per Specification 3.2.3 within the next 1 hour or,

    2. Reduce THERMAL POWER to less than 30% of RATED THERMAL POWER within the next 6 hours.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within 1 hour. Otherwise, either:

TABLE 3.3.4.2-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[a]
1. Turbine Stop Valve - Closure	2[b]
2. Turbine Control Valve - Fast Closure	2[b]

6

[a] A trip system may be placed in an inoperable status for up to 2 hours for required surveillance provided that the other trip system is OPERABLE.

[b] This function shall be automatically bypassed when turbine first stage pressure is less than or equal to 140 psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER.

TABLE 4.3.4.2.1-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Turbine Stop Valve-Closure	~~M~~ → Q	R
2. Turbine Control Valve-Fast Closure	~~M~~ → Q	R

TABLE 3.3.5-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

FUNCTIONAL UNITS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM[a]	ACTION
a. Reactor Vessel Water Level - Low Low, Level 2	2	50
b. Reactor Vessel Water Level - High, Level 8	2[b]	51
c. Manual Initiation	1[c]	52

6

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

(b) One trip system with two-out-of-two logic.

(c) Single channel.

TABLE 3.3.5-1 (Continued)

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

ACTION 50 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement:

a. For one trip system, place the inoperable channel in the tripped condition within one hour or declare the RCIC system inoperable. 24 hours

b. For both trip systems, declare the RCIC system inoperable.

ACTION 51 - With the number of OPERABLE channel less than required by the minimum OPERABLE Channels per Trip System requirement, declare the RCIC system inoperable, within 24 hours.

ACTION 52 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, restore the inoperable channel to OPERABLE status within 8 hours or declare the RCIC system inoperable. 24

TABLE 4.3.5.1-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNITS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
a. Reactor Vessel Water Level - Low Low, Level 2	NA	~~M~~ → Q	R
b. Reactor Vessel Water Level - High, Level 8	NA	~~M~~ → Q	R
c. Manual Initiation	NA	R	NA

3/4.3.6 CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6 The control rod withdrawal block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

a. With a control rod withdrawal block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement, take the ACTION required by Table 3.3.6-1.

SURVEILLANCE REQUIREMENTS

4.3.6 Each of the above required control rod withdrawal block trip systems and instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST* and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.6-1.

Insert A

Insert "A" to LSCS Technical Specification 4.3.6

* A channel may be placed in an inoperable status for up to 6 hours for required surveillance (or 12 hours for repair) without placing the trip system in the tripped condition provided at least one other OPERABLE in the same trip system is monitoring that parameter.

TABLE 3.3.6-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

ACTION

ACTION 60 - Declare the RBM inoperable and take the ACTION required by Specification 3.1.4.3.

ACTION 61 - With the number of OPERABLE channels:

a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 7 days or place the inoperable channel in the tripped condition within the next hour.

b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.

ACTION 62 - With the number of OPERABLE Channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour. 12 hours.

NOTE

* With THERMAL POWER $\geq$ 30% of RATED THERMAL POWER.

** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

a. The RBM shall be automatically bypassed when a peripheral control rod is selected.

b. This function shall be automatically bypassed if detector count rate is $\geq$ 100 cps or the IRM channels are on range 3 or higher.

c. This function shall be automatically bypassed when the associated IRM channels are on range 8 or higher.

d. This function shall be automatically bypassed when the IRM channels are on range 3 or higher.

e. This function shall be automatically bypassed when the IRM channels are on range 1.

TABLE 4.3.6-1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	NA	S/U[b][c], ~~M~~[c] Q	Q	1*
b. Inoperative	NA	S/U[b][c], ~~M~~[c] Q	N.A.	1*
c. Downscale	NA	S/U[b][c], ~~M~~[c] Q	Q	1*
2. APRM				
a. Flow Biased Simulated Thermal Power-Upscale	NA	S/U[b], ~~M~~ Q	SA	1
b. Inoperative	NA	S/U[b], ~~M~~ Q	N.A.	1, 2, 5
c. Downscale	NA	S/U[b], ~~M~~ Q	SA	1
d. Neutron Flux-High	NA	S/U[b], ~~M~~ Q	SA	2, 5
3. SOURCE RANGE MONITORS				
a. Detector not full in	NA	S/U[b], W	N.A.	2, 5
b. Upscale	NA	S/U[b], W	Q	2, 5
c. Inoperative	NA	S/U[b], W	N.A.	2, 5
d. Downscale	NA	S/U[b], W	Q	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	S/U[b], W	N.A.	2, 5
b. Upscale	NA	S/U[b], W	Q	2, 5
c. Inoperative	NA	S/U[b], W	N.A.	2, 5
d. Downscale	NA	S/U[b], W	Q	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	NA	Q	R	1, 2, 5**
b. Scram Discharge Volume Switch in Bypass	NA	~~M~~ Q	N.A.	5**
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	NA	S/U[b], ~~M~~ Q	Q	1
b. Inoperative	NA	S/U[b], ~~M~~ Q	N.A.	1
c. Comparator	NA	S/U[b], ~~M~~ Q	Q	1

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

a. Preserve the integrity of the fuel cladding.

b. Preserve the integrity of the reactor coolant system.

c. Minimize the energy which must be adsorbed following a loss-of-coolant accident, and

d. Prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. The system meets the intent of IEEE-279, 1971, for nuclear power plant protection systems. ~~The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1.~~ Insert A

The measurement of response time at the specified frequencies provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the accident analysis. No credit was taken for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurement, provided such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) inplace, onsite or offsite test measurements, or (2) utilizing replacement sensors with certified response times.

Insert "A" to LSCS Technical Specification 3/4.3.1

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P-A, "Technical Specification Improvement Analysis for BWR Reactor Protection System," March 1988.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

This specification ensures the effectiveness of the instrumentation used to mitigate the consequences of accidents by prescribing the OPERABILITY trip setpoints and response times for isolation of the reactor systems. When necessary, one channel may be inoperable for brief intervals to conduct required surveillance.

Insert (A) (New Paragraph)

Some of the trip settings may have tolerances explicitly stated where both the high and low values are critical and may have a substantial effect on safety. The setpoints of other instrumentation, where only the high or low end of the setting have a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

Except for the MSIVs, the safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. For D.C. operated valves, a 3 second delay is assumed before the valve starts to move. For A.C. operated valves, it is assumed that the A.C. power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of 13 seconds is assumed before the valve starts to move. In addition to the pipe break, the failure of the D.C. operated valve is assumed; thus the signal delay is concurrent with the 13 second diesel startup. The safety analysis considers an allowable inventory loss in each case which in turn determines the valve speed in conjunction with the 13 second delay. It follows that checking the valve speeds and the 13 second time for emergency power establishment will establish the response time for the isolation functions. However, to enhance overall system reliability and to monitor instrument channel response time trends, the isolation actuation instrumentation response time shall be measured and recorded as a part of the ISOLATION SYSTEM RESPONSE TIME.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

The emergency core cooling system actuation instrumentation is provided to initiate actions to mitigate the consequences of accidents that are beyond the ability of the operator to control. This specification provides the OPERABILITY requirements, trip setpoints and response times that will ensure effectiveness of the systems to provide the design protection. Although the instruments are listed by system, in some cases the same instrument may be used to send the actuation signal to more than one system at the same time.

Insert (B)

Insert "A" to LSCS Technical Specification 3/4.3.2

Specified surveillance intervals and maintenance outage times have been determined in accordance with NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation", June 1990.

Insert "B" to LSCS Technical Specification 3/4.3.3

Specified surveillance intervals and maintenance outage times have been determined in accordance with NEDC-30936P-A, Parts 1 and 2, "BWR Owners' Group Technical Specification Improvement Methodology (with Demonstration for BWR ECCS Actuation Instrumentation)", December 1988, plus letter OG90-319-32D to M. L. Wohl from W. P. Sullivan dated March 22, 1990, "Clarification of Technical Specification Changes Given in ECCS Actuation Instrumentation Analysis".

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION

The anticipated transient without scram (ATWS) recirculation pump trip system provides a means of limiting the consequences of the unlikely occurrence of a failure to scram during an anticipated transient. The response of the plant to this postulated event falls within the envelope of study events in General Electric Company Topical Report NEDO-10349, dated March 1971 and NEDO-24222, dated December, 1979, and Appendix G of the FSAR.

The end-of-cycle recirculation pump trip (EOC-RPT) system is a part of the Reactor Protection System and is an essential safety supplement to the reactor trip. The purpose of the EOC-RPT is to recover the loss of thermal margin which occurs at the end-of-cycle. The physical phenomenon involved is that the void reactivity feedback due to a pressurization transient can add positive reactivity to the reactor system at a faster rate than the control rods add negative scram reactivity. Each EOC-RPT system trips both recirculation pumps, reducing coolant flow in order to reduce the void collapse in the core during two of the most limiting pressurization events. The two events for which the EOC-RPT protective feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

A generic analysis, which provides for continued operation with one or both trip systems of the EOC-RPT system inoperable, has been performed. The analysis determined bounding cycle independent MINIMUM CRITICAL POWER RATIO (MCPR) Limiting Condition for Operation (LCO) values which must be used if the EOC-RPT system is inoperable. These values ensure that adequate reactivity margin to the MCPR safety limit exists in the event of the analyzed transient with the RPT function inoperable. The analysis results are further discussed in the bases for Specification 3.2.3.

A fast closure sensor from each of two turbine control valves provides input to the EOC-RPT system; a fast closure sensor from each of the other two turbine control valves provides input to the second EOC-RPT system. Similarly, a position switch for each of two turbine stop valves provides input to one EOC-RPT system; a position switch from each of the other two stop valves provides input to the other EOC-RPT system. For each EOC-RPT system, the sensor relay contacts are arranged to form a 2-out-of-2 logic for the fast closure of turbine control valves and a 2-out-of-2 logic for the turbine stop valves. The operation of either logic will actuate the EOC-RPT system and trip both recirculation pumps.

Insert A

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 30% of RATED THERMAL POWER are annunciated in the control room.

The EOC-RPT system response time is the time assumed in the analysis between initiation of valve motion and complete suppression of the electric arc, i.e., 190 ms, less the time allotted for sensor response, i.e., 10 ms, and less the time allotted for breaker arc suppression determined by test, as correlated to manufacturer's test results, i.e., 83 ms, and plant pre-operational test results.

Insert "A" to LSCS Technical Specification 3/4.3.4

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with NEDC-30851P-A, "Technical Specification Improvement Analysis for BWR Reactor Protection System," March 1988.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without providing actuation of any of the emergency core cooling equipment.

Insert (A) →

3/4.3.6 CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of the specifications in Section 3/4.1.4, Control Rod Program Controls and Section 3/4.2 Power Distribution Limits. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

Insert (B) →

3/4.3.7 MONITORING INSTRUMENTATION

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring instrumentation ensures that; (1) the radiation levels are continually measured in the areas served by the individual channels, and (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3.4.3.7.2 SEISMIC MONITORING INSTRUMENTATION

The OPERABILITY of the seismic monitoring instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the unit. This instrumentation is consistent with the recommendations of Regulatory Guide 1.12 "Instrumentation for Earthquakes", April 1974.

3/4.3.7.3 METEOROLOGICAL MONITORING INSTRUMENTATION

The OPERABILITY of the meteorological monitoring instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public. This instrumentation is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," February, 1972.

3/4.3.7.4 REMOTE SHUTDOWN MONITORING INSTRUMENTATION

The OPERABILITY of the remote shutdown monitoring instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN of the unit from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

Insert "A" to LSCS Technical Specification 3/4.3.5

Specified surveillance intervals and maintenance outage times have been determined in accordance with NEDC-30936P-A, Parts 1 and 2, "BWR Owners' Group Technical Specification Improvement Methodology (with Demonstration for BWR ECCS Actuation Instrumentation)", December 1988 plus letter OG90-319-32D to M. L. Wohl from W. P. Sullivan dated March 22, 1990, "Clarification of Technical Specification Changes Given in ECCS Actuation Instrumentation Analysis".

Insert "B" to LSCS Technical Specification 3/4.3.6

Specified surveillance intervals and maintenance outage times have been determined in accordance with NEDC-30851P-A, Supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation", October 1988.

ATTACHMENT L

BWR OWNERS GROUP LETTER TO B.K. GRIMES (NUCLEAR REACTOR REGULATION) BWR OWNERS GROUP (BWROG) TOPICAL REPORTS ON TECHNICAL SPECIFICATION IMPROVEMENT ANALYSIS FOR BWR REACTOR PROTECTION SYSTEMS-USE FOR RELAY AND SOLID STATE PLANTS (NEDC-30884 AND NEDC-30851 P)

BWROG 92102, NOVEMBER 4, 1992

BWR OWNERS' GROUP

Cynthia L. Tully, Chairperson
(205) 877-7357

c/o Southern Nuclear Operating Company • P.O. Box 1295, Bin B052 • Birmingham, AL 35201

BWROG-92102
November 4, 1992

Brian K. Grimes, Director
Division of Operating Reactor Support
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Christopher I. Grimes, Chief
Technical Specifications Branch

Subject: **BWR OWNERS' GROUP (BWROG) TOPICAL REPORTS ON TECHNICAL SPECIFICATION IMPROVEMENT ANALYSIS FOR BWR REACTOR PROTECTION SYSTEMS - USE FOR RELAY AND SOLID STATE PLANTS (NEDC-30884 AND NEDC-30851P)**

Reference: Letter, C.E. Rossi (NRC) to G.J. Beck (BWROG), same subject, dated July 26, 1991

Dear Mr. Grimes:

In the reference letter the NRC expressed concern that model TS ACTION "a", proposed in NEDC-30851P-A, would allow continued plant operation for up to 12 hours with a combination of failures that could prevent a reactor scram function from completing its logic when called upon (i.e., loss-of-function). This could occur for a relay-type plant if, for example, both channels of the high reactor pressure function (which has a one-out-of-two-twice logic) were inoperable in one trip system. The reference letter also noted that the BWROG was preparing clarifying language (i.e., revised model TS ACTIONs) to address this concern, and to be used as an industry standard in future amendments implementing the RPS topical report.

In response to the above, the BWROG has worked with Carl Schulten of your Staff to develop the model TS ACTIONs provided in Enclosure 1. The indicated changes to ACTIONs 3.3.1a and 3.3.1b and their footnotes ensure that appropriate actions are taken to avoid an extended loss-of-function period in any RPS Functional Unit. A discussion of the application and justification for the revised model TS ACTIONs is provided in Enclosure 2.

The enclosed information has been endorsed by a substantial number of the members of the BWROG; however, it should not be interpreted as a commitment of any individual member to a specific course of action. Each member must formally endorse the BWROG position in order for that position to become the member's position.

Very truly yours,

Cynthia L. Tully

C. L. Tully, Chairperson
BWR Owners' Group

EXEC6T/CLT/JDF/rt
Enclosures 2

cc: CS Schulten (NRC)
LA England, BWROG Vice Chairman
BWROG Primary Representatives of Participating Utilities
BWROG Technical Specifications Committee-D
LS Gifford, GE
SJ Stark, GE

Table 5-9

CHANGES TO RELAY RPS TECHNICAL SPECIFICATION

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION: INSERT 1

~~a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel(s) and/or that trip system in the tripped condition* within twelve hours. The provisions of Specification 3.0.4 are not applicable.~~

~~b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within one hour and take the ACTION required by Table 3.3.1-1.~~

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

INSERT 2

~~*An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 6 hours or the ACTION required by Table 3.3.1-1 for that Trip Function shall be taken.~~

~~**If more channels are inoperable in one trip system than in the other, place the trip system with more inoperable channels in the tripped condition, except when this would cause the Trip Function to occur.~~

INSERT 1:

a. With one channel required by Table 3.3.1-1 inoperable in one or more Functional Units, place the inoperable channel and/or that trip system in the tripped condition* within 12 hours. [The provisions of Specification 3.0.4 are not applicable.]

b. With two or more channels required by Table 3.3.1-1 inoperable in one or more Functional Units:

   1. Within one hour, verify sufficient channels remain OPERABLE or tripped* to maintain trip capability in the Functional Unit, and

   2. Within 6 hours, place the inoperable channel(s) in one trip system and/or that trip system** in the tripped condition*, and

   3. Within 12 hours, restore the inoperable channels in the other trip system to an OPERABLE status or tripped*.

   Otherwise, take the ACTION required by Table 3.3.1-1 for the Functional Unit.

INSERT 2:

*An inoperable channel or trip system need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, if the inoperable channel is not restored to OPERABLE status within the required time, the ACTION required by Table 3.3.1-1 for the Functional Unit shall be taken.

**This ACTION applies to that trip system with the most inoperable channels; if both trip systems have the same number of inoperable channels, the ACTION can be applied to either trip system.

<u>Application and Justification for Changes to NEDC-30851P-A, Table 5-9</u>

For ACTION 3.3.1a, with one channel required by Table 3.3.1-1 inoperable in one or more Functional Unit(s) (i.e., any number of Functional Units having only one inoperable channel in each Functional Unit), the entire RPS scram capability remains intact, assuming no additional single failure. Therefore, a loss-of-function is not possible for the rewritten ACTION 3.3.1.a. The action that allows continued operation for 12 hours was evaluated and the reliability of the system shown to be acceptable in NEDC-30851P-A.

Within 12 hours the inoperable channels and/or trip system must be placed in the tripped condition. This action restores the RPS capability to accommodate a single failure and allows operation to continue with no further restrictions. If the inoperable channel(s) and/or trip system is not placed in the tripped condition within the required time (12 hours for ACTION a), then the ACTIONs required by Table 3.3.1-1 must be taken, which require the operators to take actions to compensate for the inoperable RPS channels' function.

For ACTION 3.3.1b, with two or more channels inoperable in any Functional Unit, the Reactor Protection System <u>may</u> not be capable of performing its intended function (i.e., a "loss of scram function" may exist), depending on which two (or more) channels are inoperable. In this condition, during the period allowed to place the inoperable channels and/or trip system in the tripped condition, if a valid trip signal was received, a failure to produce a scram signal for that Functional Unit could result. Therefore, ACTION b.1 requires that steps be taken within one hour to ensure the Functional Unit maintains trip capability. This one hour period allows the operator time to evaluate the situation and to repair or trip the channels. One hour is reasonable considering the diversity of sensors available to provide trip signals, and the low probability of an event requiring the initiation of a scram. One hour is also consistent with the current Technical Specification requirement for placing inoperable channels in the tripped condition. In addition, if it has been verified that a loss-of-function situation does not exist, an allowance of 6 hours is provided by ACTION b.2 before the operator is required to place the inoperable channel(s) in one Trip System (or one entire Trip System), in the tripped condition. This 6 hour requirement limits the time the RPS scram logic for any Functional Unit may be degraded in both Trip Systems. Six hours is considered acceptable based on the remaining capability to trip, the diversity available to provide trip signals, the low probability of extensive numbers of inoperabilities affecting all diverse functions, and the low probability of an event requiring the initiation of a scram. By the end of the six hour period, the ACTION b.2 requirement that one Trip System will have its inoperable channels placed into the tripped condition provides a similar level of RPS availability as found in ACTION a above, and evaluated in NEDC-30851P-A to be acceptable for a 12 hour allowable outage time.

Within 12 hours, per ACTION b.3, all the inoperable channels in the other trip system will have been restored to OPERABLE status, or else the inoperable channels will be placed in trip. For all of the proposed ACTIONs, if the inoperable channels are not placed in trip within the applicable required time (1, 6, or 12 hours), then the ACTIONs required by Table 3.3.1-1 must be taken, which requires the operators to take actions to compensate for the inoperable RPS channels' function.