12. (NPE-85/06)

SUBJECT: Technical Specifications Tables 3.3.7.4-1, 3.4.3.2-1, 3.6.4-1, 3.8.4.1-1 and 3.8.4.2-1, pages 3/4 3-71, 3/4 4-11, 3/4 6-30, 3/4 6-31, 3/4 6-38, 3/4 8-32, and 3/4 8-47

DISCUSSION:

This technical specification change results in part from a design change to add a more accessible RHR to Head Spray containment isolation valve and is planned for completion during an outage currently scheduled to begin in October, 1985.

It is proposed to designate valve E12-F394 as the RHR to Head Spray Reactor Coolant System Pressure and Containment Isolation Valve and no longer classify valve E51-F066 as the inboard isolation valve. A footnote requiring any required revision of the proposed 35 second closure time be submitted within 90 days after completion of the second ASME Section XI closure time test is also proposed. It is further proposed that valve E12-F344 be deleted from the Containment and Drywell Isolation Valve Table 3.6.4-1 and that the penetration protection circuit breaker and the thermal overload protection associated with the power circuitry for valve E12-F394 be added to tables 3.8.4.1-1 and 3.8.4.2-1, respectively.

It is also proposed to make an editorial clarification change to ensure that the technical specifications reflect the correct nomenclature for the following valves:

Page 3/4 3-71, Items 26 and 27; Reactor Core Isolation Cooling Test Return to Condensate Storage Tank Inboard, and Outboard, Valves ("RCIC Test RTN to CST IB (OB) Valve"), respectively.

Page 3/4 6-31, Valves E12-F028A-A and B-B; Residual Heat Removal Heat Exchanger "A" ("B") to Containment Spray Sparger Inlet ("RHR Heat Exchanger "A" ("B") to CTMT SPR Sparger INL"), and for valves E12-F037A-A and B-B, "RHR Heat Exchanger "A" ("B") to CTMT Pool."

(It should be noted that attached technical specification page 3/4 6-31 reflects a previously requested change, NLS-85/06, which was transmitted as Item 4 of MP&L serial AECM-85/0168 on July 3, 1985, and that attached technical specification page 3/4 4-11 reflects change NPE-85/15 which is requested by item 13 of this submittal.)

JUSTIFICATION:

Local leak rate testing of the current RHR to Head Spray isolation valve E51-F066 requires removal of the drywell head and insulation because of the normally inaccessible location of the valve. To enhance operational effectiveness, a design change has been performed which added motor operated valve E12-F394 in a more accessible location upstream of valve E51-F066. The proposed technical specification change designates E12-F394 as the RHR to Head Spray isolation valve instead of E51-F066.

Since E12-F394 is upstream of the test connection which valve E12-F344 isolates, the test connection is no longer a leakage path from the drywell and it is also proposed to delete valve E12-F344 from table 3.6.4-1.

Valve E12-F394 does not have an analytical closure time defined in the FSAR accident analysis. The closure time for this valve must therefore be determined from ASME Section XI test data as described in Technical Specification Bases 3/4.6.4. Since the valve cannot be made operable during power operation until after approval of this proposed change, this data is not yet available. The closure time is therefore qualified with a footnote that any required change to the closure time be submitted within 90 days after completion of the second ASME Section XI closure time test. The proposed closure time of 35 seconds was chosen as a reasonable interim closing time because valve G33-F253, which is of the same manufacture and was procured through very similar purchase specifications, is currently listed in the technical specifications and has been found to have a maximum closure time by technical specification bases criteria of 35 seconds. It is expected that valve E12-F394 will exhibit closure characteristics similar to those of valve G33-F253.

The addition of the appropriate penetration protection circuit breaker and the thermal overload protection to tables 3.8.4.1-1 and 3.8.4.2-1 associated with valve E12-F394 will make the technical specification consistent with the safety analysis and the as-built plant when the electrical connections are terminated after approval of this proposed change.

This design change as proposed adds a local handswitch on the remote shutdown panel for valve E12-F394. MP&L does not propose to add this handswitch to Technical Specification 3/4.3.7.4. Neither the GGNS safety analysis nor the GGNS Fire Protection Plan requires operability of the E12-F394 valve from the remote shutdown panel for any analyzed condition. It should be further noted that while the design change includes an interlock such as is presently included for valve E12-F023 to prevent inadvertant overpressurization of the RHR system through the head spray line, this interlock will not be included in the remote shutdown panel control circuit. Overpressurization of the RHR system through the head spray line from the remote shutdown panel would require the following sequence of events:

1. A control room fire or similar circumstances which would render the main control room uninhabitable, thereby requiring operation from the remote shutdown panel,

2. an operator error resulting in the opening of both motor operated valves E12-F023 and E12-F394 against a reactor pressure which is greater than the design pressure of the RHR system, and

3. failure of the outboard check valve E12-F019.

The foregoing is considered to be an incredible sequence of events involving a minimum of three single failures, and incorporation of a pressure interlock for the remote shutdown system control circuit is therefore not required.

The changes correcting the valve nomenclatures are editorial changes proposed solely to enhance the clarity of the subject tables. They will ensure that the technical specifications reflect the specific functions of these valves and will make the technical specifications consistent with the specific nomenclature used in the as-built plant.

SIGNIFICANT HAZARDS CONSIDERATION:

The proposed changes are operational enhancements which will decrease operational down time during leak rate testing since it will no longer require removal of the drywell head and insulation to accomplish the testing. These changes will also reduce personnel exposure since radiation workers will not be required to remain in the radiation area for extended periods to accomplish the leak rate testing.

The addition of valve E12-F394 to the RHR Head Spray line and the installation of its attendant hardware were performed in accordance with applicable industry and regulatory codes and standards and the GGNS Quality Assurance Program, and as such are consistent with the licensing bases and the safety analyses. The additional changes requested by this item are purely administrative in nature in that they are corrections of valve nomenclature only. The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated or create the possibility of a new or different kind of accident from any accident previously evaluated, nor do they involve a significant reduction in a margin of safety.

Therefore, the proposed changes involve no significant hazards considerations.

12. (NPE-85/06)

TABLE 3.3.7.4-1 (Continued)

REMOTE SHUTDOWN SYSTEM CONTROLS

CONTROL	MINIMUM CHANNELS OPERABLE Div 1	Div 2
12. RHR Injection Valves	2^b	2^b
13. RHR Test Line Valve	1	1
14. RHR HX Cond. to RCIC Valve	1	1
15. RHR HX Flow to Suppression Pool Valve	1	1
16. RHR Discharge to Radwaste Valve	1	1
17. RCIC Steam to RHR HX Valve	2^b	2^b
18. Diesel Generator HX Inlet Valve	1	1
19. Safety/Relief Valves	6^b	6^b
20. RHR to RCIC Head Spray Line Valve	1	NA
21. RCIC Turbine Flow Controller	1	NA
22. RCIC Suction Flow Suppression Pool Valve	1	NA
23. RCIC Injection Shutoff Valve	1	NA
24. RCIC Suction From CST	1	NA
25. RCIC Recirc. Main Flow Bypass Valve	1	NA
26. RCIC Test ~~FCV~~ RTN to CST IB Valve	1	NA
27. RCIC Test RTN to CST OB Valve	1	NA
28. Steam to RCIC Turbine Valve	1	NA
29. RCIC Turbine Trip & Throttle Valve	1	NA
30. RCIC Turbine Cooling Water Valve	1	NA
31. RCIC Turbine Local Control Select Switch	1	NA
32. RCIC Gland Seal Compressor	1	NA
33. Shutdown Cooling Isolation Valve Reset Switch	1	1

NOTE: a. 1 per cooling tower fan
b. 1 per valve

TABLE 3.4.3.2-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	SYSTEM
E21-F005 E21-F006	LPCS
E22-F004 E22-F005	HPCS
E12-F008 E12-F009 E12-F023 E12-F041 A, B, C E12-F042 A, B, C E12-F050 A, B E12-F053 A, B E12-F308 E12-F394	RHR
E51-F063 E51-F064 E51-F065 ~~E51-F066~~ E51-F076 E51-F013	RCIC

TABLE 3.4.3.2-2

REACTOR COOLANT SYSTEM INTERFACE VALVES PRESSURE MONITORS - ALARM

VALVE NUMBER	SYSTEM	ALARM SETPOINT (psig)
E21-F005 to E21-F006	LPCS	~~≤ 50~~ 575
E12-F008 to E12-F006A	RHR	≤ 183
E12-F008 to E12-F006B	RHR	≤ 183
E12-F041A to E12-F042A	RHR	~~≤ 50~~ 475
E12-F041B to E12-F042B	RHR	~~≤ 50~~ 475
E12-F041C to E12-F042C	RHR	~~≤ 50~~ 475

(NPE-85/15)

TABLE 3.6.4-1
CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM AND VALVE NUMBER		PENETRATION NUMBER	VALVE GROUP[(a)]	MAXIMUM ISOLATION TIME (Seconds)
1. Automatic Isolation Valves#				
a. Containment				
Main Steam Lines	B21-F028A	5(0)*	1	5
Main Steam Lines	B21-F022A	5(I)*	1	5
Main Steam Lines	B21-F067A-A	5(0)*	1	9
Main Steam Lines	B21-F028B	6(0)*	1	5
Main Steam Lines	B21-F022B	6(I)*	1	5
Main Steam Lines	B21-F067B-A	6(0)*	1	9
Main Steam Lines	B21-F028C	7(0)*	1	5
Main Steam Lines	B21-F022C	7(1)*	1	5
Main Steam Lines	B21-F067C-A	7(0)*	1	9
Main Steam Lines	B21-F028D	8(0)*	1	5
Main Steam Lines	B21-F022D	8(I)*	1	5
Main Steam Lines	B21-F067D-A	8(0)*	1	9
RHR Reactor Shutdown Cooling Suction	E12-F008-A	14(0)	3	40
RHR Reactor Shutdown Cooling Suction	E12-F009-B	14(I)	3	40
Steam Supply to RHR and RCIC Turbine	E51-F063-B	17(I)	4	20
Steam Supply to RHR and RCIC Turbine	E51-F064-A	17(0)	4	20
Steam Supply to RHR and RCIC Turbine	E51-F076-B	17(I)	4	20
RHR to Head Spray	E12-F023-A	18(0)	3	94##
RHR to Head Spray	E12-F394-B	18(I)	3	35
Main Steam Line Drains	B21-F019-A	19(0)	1	20

(a) See Specification 3.3.2, Table 3.3.2-1, for isolation signal(s) that operates each valve group.
(b) Deleted
(c) Hydrostatically tested with water to 1.10 P_a, 12.65 psig.
(d) Hydrostatically tested by pressurizing system to 1.10 P_a, 12.65 psig.
(e) Hydrostatically tested during system functional tests.
(f) Deleted
(g) Normally closed or locked closed manual valves may be opened on an intermittent basis under administrative control.

*The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITONS 2 or 3 provided the surveillance is performed within 12 hours after reaching a reactor steam pressure of 600 psig and prior to entry into OPERATIONAL CONDITION 1.

#The "-A, -B, -C, -(A), -(B), -(C)" designators on the valve numbers indicate associated electrical divisions.

(Insert)

12. (NPE-85/06)

INSERT to page 3/4 6-30

Initial closure time. Final closure time to be determined during ASME Section XI testing. Any required change to this closure time shall be submitted to the Commission within 90 days of the second closure time test completion.

12. (NPE-85/06)

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM AND VALVE NUMBER		PENETRATION NUMBER	VALVE GROUP[(a)]	MAXIMUM ISOLATION TIME (Seconds)
Containment (Continued)				
Main Steam Line Drains	B21-F016-B	19(I)	1	20
RHR Heat Exchanger "A" to ~~LPCI~~ CTMT SPR Sparger INL	E12-F028A-A	20(I)	5	90
RHR Heat Exchanger "A" to ~~LPCI~~ CTMT Pool	E12-F037A-A	20(I)	3	74
RHR Heat Exchanger "B" to ~~LPCI~~ CTMT SPR Sparger INL	E12-F028B-B	21(I)	5	90
RHR Heat Exchanger "B" to ~~LPCI~~ CTMT Pool	E12-F037B-B	21(I)	3	74
RHR "A" Test Line to Supp. Pool	E12-F024A-A	23(O)[(d)]	5	90
RHR "A" Test Line to Supp. Pool	E12-F011A-A	23(O)[(d)]	5	36
RHR "C" Test Line to Supp. Pool	E12-F021-B	24(O)[(d)]	5	144
HPCS Test Line	E22-F023-C	27(O)[(d)]	6B	75
RCIC Pump Suction	E51-F031-A	28(O)[(d)]	4	56
RCIC Turbine Exhaust	E51-F077-A	29(O)[(c)]	9	26
LPCS Test Line	E21-F012-A	32(O)[(d)]	5	144
Cont. Purge and Vent Air Supply	M41-F011-(A)	34(O)	7	4
Cont. Purge and Vent Air Supply	M41-F012-(B)	34(I)	7	4
Cont. Purge and Vent Air Exh.	M41-F034-(B)	35(I)	7	4
Cont. Purge and Vent Air Exh.	M41-F035-(A)	35(O)	7	4
Plant Service Water Return	P44-F070-B	36(I)	6A	33
Plant Service Water Return	P44-F069-A	36(O)	6A	33
Plant Service Water Supply	P44-F053-A	37(O)	6A	33
Chilled Water Supply	P71-F150-(A)	38(O)	6A	12

NLS-85/06

12.(NPE-85/06)

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM AND VALVE NUMBER		PENETRATION NUMBER
b. Drywell		
Cont. Cooling Water Inlet	P42-F114-B	329(O)
Cont. Cooling Water Outlet	P42-F116-A	330(I)
Cont. Cooling Water Outlet	P42-F117-B	330(O)
3. Other Isolation Valves(g)#		
a. Containment		
Fuel Transfer Tube	F11-E015	4(I)
Feedwater Inlet	B21-F010A	9(I)
Feedwater Inlet	B21-F032A	9(O)
Feedwater Inlet	B21-F010B	10(I)
Feedwater Inlet	B21-F032B	10(O)
RHR "A" Suction	E12-F017A	11(O)(d)
RHR "B" Suction	E12-F017B	12(O)(d)
RHR "C" Suction	E12-F017C	13(O)(d)
RHR Shutdown Cooling Suction	E12-F308	14(I)
~~RHR Head Spray~~	~~E51-F066-(A)~~	~~18(I)~~
~~RHR Head Spray~~	~~E12-F344~~	~~18(I)~~
RHR Heat Ex. "A" to LPCI	E12-F044A	20(I)
RHR Heat Ex. "A" to LPCI	E12-F025A	20(I)
RHR Heat Ex. "A" to LPCI	E12-F107A	20(I)
RHR Heat Ex. "B" to LPCI	E12-F025B	21(I)
RHR Heat Ex. "B" to LPCI	E12-F044B	21(I)
RHR Heat Ex. "B" to LPCI	E12-F107B	21(I)
RHR Heat Ex. "C" to LPCI	E12-F234	22(O)
RHR Pump "C" to LPCI	E12-F041C-B	22(I)
RHR Pump "A" Test Line to Suppr. Pool	E12-F259	23(O)(e)
RHR Pump "A" Test Line to Suppr. Pool	E12-F261	23(O)(e)

TABLE 3.8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

c. 480 VAC Circuit Breakers (Continued)

Molded Case, Type NZM

BREAKER NUMBER	TRIP SETPOINT (Amperes)	RESPONSE TIME (Seconds)	SYSTEM/COMPONENT AFFECTED
52-1542-29	1200	0.100	STBY LIQ CONTROL SYS MIXING HEATER (Q1C41D003)
52-1611-10	12.5	0.100	MOV - DRYWELL COLL TK OUTLET ISOLATION (Q1G41F044-B)
52-1611-15	12.5	0.100	MOV - PSW CTMT STM TNL CLR ISOL (Q1P44F070-B)
52-1611-16	50	0.100	MOV-RHR RX HD SPR INBD ISOL (Q1E12F394-B)
52-1611-25	12.5	0.100	MOV - DRYWELL CLG WTR ISOL (Q1P42F117-B)
52-1611-31	12.5	0.100	MOV - DRYWELL CLG WTR INL ISOL (Q1P42F114-B)
52-1611-32	32	0.100	MOV - CTMT CLG WTR ISOLATION (Q1P42F068-B)
52-1611-42	12.5	0.100	MOV PSW STEAM TUNNEL CLR ISOL (Q1P44F074-B)
52-1611-43	12.5	0.100	MOV PSW STEAM TUNNEL CLR ISOL (Q1P44F077-B)
52-1611-44	38	0.100	MOV - SERVICE AIR DRYWELL ISOLATION (Q1P52F195-B)
52-1621-03	7	0.100	MOV - DRWL HYDR INST LINE ISO (Q1E61F595B-B)
52-1621-04	7	0.100	MOV - DRWL HYDR INST LINE ISO (Q1E61F597B-B)

TABLE 3.8.4.2-1

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	BYPASS DEVICE (CONTINUOUS) (ACCIDENT CONDITIONS) (NO)	SYSTEM(S) AFFECTED
Q1E51F010	Continuous	RCIC System
Q1E51F013	Continuous	RCIC System
Q1E51F019	Continuous	RCIC System
Q1E51F022	Continuous	RCIC System
Q1E51F031	Continuous	RCIC System
Q1E51F045	Continuous	RCIC System
Q1E51F046	Continuous	RCIC System
Q1E51F059	Continuous	RCIC System
Q1E51F068	Continuous	RCIC System
RCIC Trip and Throttle Valve on Turbine Q1E51C002	Continuous	RCIC System
Q1B21F065A	No	Reactor Coolant System
Q1B21F065B	No	Reactor Coolant System
Q1B21F098A	No	Reactor Coolant System
Q1B21F098B	No	Reactor Coolant System
Q1B21F098C	No	Reactor Coolant System
Q1B21F098D	No	Reactor Coolant System
Q1B21F019	Continuous	Reactor Coolant System
Q1B21F067A	Continuous	Reactor Coolant System
Q1B21F067B	Continuous	Reactor Coolant System
Q1B21F067C	Continuous	Reactor Coolant System
Q1B21F067D	Continuous	Reactor Coolant System
Q1B21F016	Continuous	Reactor Coolant System
Q1B21F147A	Continuous	MSL Drain Post LOCA Leakage Control
Q1B21F147B	Continuous	MSL Drain Post LOCA Leakage Control
Q1B33F019	Continuous	Recirculation System
Q1B33F020	Continuous	Recirculation System
Q1B33F125	Continuous	Recirculation System
Q1B33F126	Continuous	Recirculation System
Q1B33F127	Continuous	Recirculation System
Q1B33F128	Continuous	Recirculation System
Q1D23F591	*	Drywell Monitoring System
Q1D23F592	*	Drywell Monitoring System
Q1D23F593	*	Drywell Monitoring System
Q1D23F594	*	Drywell Monitoring System
Q1E12F040	Continuous	RHR System
Q1E12F023	Continuous	RHR System
Q1E12F006A	Continuous	RHR System
Q1E12F052A	Continuous	RHR System
Q1E12F008	Continuous	RHR System
Q1E12F394	Continuous	RHR System

13. (NPE-85/15)

SUBJECT: Technical Specification Tables 3.3.3-1, 3.3.3-2, 3.3.3-3, 4.3.3.1-1, Technical Specification 4.4.3.2.2.b, Tables 3.4.3.2-2, and 3.4.3.2-3, pages 3/4 3-28, 30, -31, -33, -34, 3/4 4-10, -11, and -12

DISCUSSION: This technical specification change results from a design change to add high/low pressure interlocks to the injection valves on the low pressure ECCS systems and is planned for implementation during an outage scheduled to begin in October, 1985.

It is proposed to add appropriate requirements for the interlocks to the subject specifications as follows:

1. Require three (3) minimum operable channels and action 31 for operational conditions 1, 2, and 3, and action 35 (proposed) for operational conditions 4* and 5* (Table 3.3.3-1, pages 3/4 3-28 and -30).

2. Require a trip setpoint and allowable value of 516 psig, decreasing and 452-534 psig, decreasing, respectively (Table 3.3.3-2, page 3/4 3-31).

3. Delete ECCS System Response Times for LPCS and LPCI mode of RHR A, B and C (Table 3.3.3-3, items 1 and 2, page 3/4 3-33).

4. Require ECCS actuation instrumentation surveillance intervals of once per: twelve hours (S) for channel check, month (M) for channel functional test, and refueling ($R^{(a)}$) for channel calibration in operational conditions 1, 2, 3, 4* and 5* (Table 4.3.3.1-1, page 3/4 3-34).

5. Delete the paragraph which implemented special requirements for the LPCS and LPCI check valves until completion of the proposed modification (Technical Specification 4.4.3.2.2.b, page 3/4 4-10).

6. Revise the alarm and interlock setpoints from 50 to 575 psig for the LPCS valve and from 50 to 475 psig for the LPCI valves (Tables 3.4.3.2-2 and -3, pages 3/4 4-11 and -12).

(It should be noted that attached technical specification page 3/4 4-11 reflects change NPE-85/06 which is requested by item 12 of this transmittal.)

JUSTIFICATION: Facility Operating License NPF-29, Condition 2.C.(18), requires MP&L to implement isolation protection against overpressurization of the low pressure emergency core cooling systems (RHR/LPCI and LPCS) prior to startup following the first refueling outage due to concerns about postulated intersystem LOCA. Completion of this design change will ensure compliance with the license condition.

The effect of adding a pressure permissive interlock to the LPCS and LPCI systems was evaluated for the DBA recirculation suction line break with failure of the LPCI diesel generator. This break and failure combination was identified in Section 6 of the GGNS FSAR as the limiting event (LOCA). The DBA suction break is most sensitive to this design change because of its rapid depressurization rate. Smaller breaks depressurize more slowly and are less affected by addition of the pressure permissive. The primary effect of the pressure permissive logic on this analyzed event is to delay the time at which the LPCS and LPCI systems begin injecting coolant into the reactor pressure vessel.

A new analysis of this postulated accident (using the existing model) was performed in which LPCI and LPCS injection was delayed by the pressure permissive interlock. The new analysis, based on the conservative assumption that LPCI and LPCS flow does not begin until the injection valves are fully open at approximately 58 seconds into the event, showed an increase in PCT of 51°F for a maximum PCT of 2149°F. Since the new PCT is still below the 10CFR50.46 limit of 2200°F, no change in the MAPLHGR limit is required.

The proposed addition of requirements to the technical specifications to support this design change will ensure appropriate surveillance of the valve interlocks. The changes result from analyses by the NSSS vendor, General Electric, and include adequate conservatism to ensure the requirements will support continued safe operation.

Since the low pressure ECCS response times are now dependent on vessel depressurization time, which is different for each type of accident analysis, the technical specification definition of ECCS system response time is not applicable to these low pressure ECCS systems. The response time of 40 seconds is no longer applicable and should be deleted from the technical specifications. Diesel generator start times (which are included in the present 40 second ECCS system response times) are tested by other technical specification requirements. The maximum valve opening time of 30 seconds assumed in the analysis will be assured by the GGNS Inservice Testing Program.

The proposed minimum operable channels requirements of three per trip function are applicable to the "one-out-of-two twice" logic incorporated into the design change, and are adequate to ensure operability of the required low pressure injection functions considering the diversity of injection systems and logic channels available (four per trip function).

The proposed allowable value for the automatic interlocks includes an allowance for uncertainty and calibration inaccuracy and the proposed trip setpoint includes an allowance for instrument drift that will ensure the rated pressure of the low pressure piping will not be exceeded and that the assumptions used in the safety analysis will be met.

The proposed surveillance frequencies are consistent with those for similar instrumentation already included in the technical specifications and are adequate to ensure required availability of the instrumentation.

The special requirements for the LPCS and LPCI check valves incorporated in Specification 4.4.3.2.2.b are no longer required and should be deleted.

The revisions to the alarm setpoints in Table 3.4.3.2-2 and the interlocks to prevent manual opening of the valves in Table 3.4.3.2-3 are consistent with the additional proposed changes, and will serve to prevent inadvertent manual overpressurization of the low pressure piping from the normal control circuits. It should be noted that the manual opening interlock setpoint for LPCS is greater than the automatic pressure permissive setpoint. This is because the pressure permissive setpoint was established to meet the safety analysis assumptions, which assume a setpoint well below that required to protect the pressure integrity of the piping. The greater manual opening interlock setpoint is acceptable to prevent exceeding the pressure rating of the LPCS piping during manual operation of the valves.

It should be further noted that consideration was given to including manual interlocks in the remote shutdown panel control circuits. Overpressurization of these systems from the remote shutdown panel would require the following sequence of events:

1. A control room fire or similar circumstances which would render the main control room uninhabitable, thereby requiring operation from the remote shutdown panel,

2. an operator error resulting in the opening of the associated motor operated valves against a reactor pressure which is greater than the design pressure of the associated system, and

3. failure of the associated system's outboard check valve.

The foregoing is considered to be an incredible sequence of events involving a minimum of three single failures, and incorporation of pressure interlocks for the remote shutdown system control circuits is therefore not required.

SIGNIFICANT HAZARDS CONSIDERATION:

The design change will be performed in accordance with appropriate regulatory and industry codes and standards, the GGNS Quality Assurance Program and the applicable requirements of the GGNS FSAR. The design change is therefore consistent with the licensing basis and the new safety analysis, which will be incorporated into the FSAR at the next annual update pending approval of these proposed changes. While the new analysis shows the design change will cause an increase of PCT to 2149°F during accident conditions, the 10 CFR 50.46 limit of 2200°F is not exceeded so the margin of safety is not reduced.

The proposed technical specification changes that add requirements not presently included in the technical specifications are considered basically conservative changes. The change that deletes the ECCS response time requirements for these valves since they will no longer be applicable and revisions to existing applicable setpoints to make them consistent with the design change will make the affected technical specifications consistent with the plant as modified by the proposed design changes. It should also be noted that these changes are responsive to NRC concerns about the prevention of intersystem LOCA.

For the reasons cited, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated or create the possibility of a new or different kind of accident from any accident previously evaluated, not do they involve a significant reduction in a margin of safety.

Therefore, the proposed changes involve no significant hazards considerations.

TABLE 3.3.3-1

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION(a)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
A. DIVISION I TRIP SYSTEM			
1. RHR-A (LPCI MODE) & LPCS SYSTEM			
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 Pump A Start Time Delay Relay	1	1, 2, 3, 4*, 5*	31
d. Manual Initiation	1/system(b)	1, 2, 3, 4*, 5*	32
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"#			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2(b)	1, 2, 3	30
b. Drywell Pressure - High	2(b)	1, 2, 3	30
c. ADS Timer	1	1, 2, 3	31
d. Reactor Vessel Water Level - Low, Level 3 (Permissive)	1	1, 2, 3	31
e. LPCS Pump Discharge Pressure-High (Permissive)	2	1, 2, 3	31
f. LPCI Pump A Discharge Pressure-High (Permissive)	2	1, 2, 3	31
g. Manual Initiation	2/system	1, 2, 3	32
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 Pump B Start Time Delay Relay	1	1, 2, 3, 4*, 5*	31
d. Manual Initiation	1/system(b)	1, 2, 3, 4*, 5*	32
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"#			
a. Reactor Vessel Water Level - Low Low Low, Level 1	2(b)	1, 2, 3	30
b. Drywell Pressure - High	2(b)	1, 2, 3	30
c. ADS Timer	1	1, 2, 3	31
d. Reactor Vessel Water Level - Low, Level 3 (Permissive)	1	1, 2, 3	31
e. LPCI Pump B and C Discharge Pressure - High (Permissive)	2/pump	1, 2, 3	31
f. Manual Initiation	2/system	1, 2, 3	32
e. Reactor Vessel Pressure - Low (Injection Permissive)	3	1,2,3	31
		4*,5*	35

13.(NPE-85/15)

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(s) inoperable.

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

ACTION 31 - 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.

ACTION 32 - 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.

ACTION 33 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel(s) in the tripped condition within one hour* or declare the HPCS system inoperable.

ACTION 34 - 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.

*The provisions of Specification 3.0.4 are not applicable.

ACTION 35 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel(s) in the tripped condition within one hour* or declare the associated system(s) inoperable.

TABLE 3.3.3-2

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
A. DIVISION 1 TRIP SYSTEM		
1. RHR-A (LPCI MODE) AND LPCS SYSTEM		
a. Reactor Vessel Water Level - Low Low Low, Level 1	$\geq$ -150.3 inches*	$\geq$ -152.5 inches
b. Drywell Pressure - High	$\leq$ 1.39 psig	$\leq$ 1.44 psig
c. LPCI Pump A Start Time Delay Relay	$\leq$ 5 seconds	$\leq$ 5.25 seconds
d. Manual Initiation	NA	NA
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"		
a. Reactor Vessel Water Level - Low Low Low, Level 1	$\geq$ -150.3 inches*	$\geq$ -152.5 inches
b. Drywell Pressure - High	$\leq$ 1.39 psig	$\leq$ 1.44 psig
c. ADS Timer	$\leq$ 105 seconds	$\leq$ 117 seconds
d. Reactor Vessel Water Level-Low, Level 3	$\geq$ 11.4 inches*	$\geq$ 10.8 inches
e. LPCS Pump Discharge Pressure-High	145 psig, increasing	125-165 psig, increasing
f. LPCI Pump A Discharge Pressure-High	125 psig, increasing	115-135 psig, increasing
g. Manual Initiation	NA	NA
B. DIVISION 2 TRIP SYSTEM		
1. RHR B AND C (LPCI MODE)		
a. Reactor Vessel Water Level - Low Low Low, Level 1	$\geq$ -150.3 inches*	$\geq$ -152.5 inches
b. Drywell Pressure - High	$\leq$ 1.39 psig	$\leq$ 1.44 psig
c. LPCI Pump B Start Time Delay Relay	$\leq$ 5 seconds	$\leq$ 5.25 seconds
d. Manual Initiation	NA	NA
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"		
a. Reactor Vessel Water Level - Low Low Low, Level 1	$\geq$ -150.3 inches*	$\geq$ -152.5 inches
b. Drywell Pressure - High	$\leq$ 1.39 psig	$\leq$ 1.44 psig
c. ADS Timer	$\leq$ 105 seconds	$\leq$ 117 seconds
d. Reactor Vessel Water Level-Low, Level 3	$\geq$ 11.4 inches*	$\geq$ 10.8 inches
e. LPCI Pump B and C Discharge Pressure-High	125 psig, increasing	115-135 psig, increasing
f. Manual Initiation	NA	NA
C. DIVISION 3 TRIP SYSTEM		
1. HPCS SYSTEM		
a. Reactor Vessel Water Level - Low Low, Level 2	$\geq$-41.6 inches*	$\geq$-43.8 inches
b. Drywell Pressure - High	$\leq$ 1.39 psig	$\leq$ 1.44 psig
c. Reactor Vessel Water Level - High, Level 8	$\leq$ 53.5 inches*	$\leq$ 55.7 inches
d. Condensate Storage Tank Level - Low	$\geq$ 0 inches	$\geq$ -3 inches
e. Suppression Pool Water Level - High	$\leq$ 5.9 inches	$\leq$ 7.0 inches
f. Manual Initiation	NA	NA
e. Reactor Vessel Pressure - Low (Injection Permissive)	516 psig, decreasing	452-534 psig, decreasing

Amendment No. —

13. (NPE-85/15)

13.(NPE-85/15)

TABLE 3.3.3-3

EMERGENCY CORE COOLING SYSTEM RESPONSE TIMES (SECONDS)

1.	LOW PRESSURE CORE SPRAY SYSTEM	~~≤ 40~~ NA
2.	LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM PUMPS A, B AND C	~~≤ 40~~ NA
3.	AUTOMATIC DEPRESSURIZATION SYSTEM	NA
4.	HIGH PRESSURE CORE SPRAY SYSTEM	≤ 27
5.	LOSS OF POWER	NA

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	S	M	R(a)	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	S	M	R(a)	1, 2, 3
c. LPCI Pump A Start Time Delay Relay	NA	M	Q	1, 2, 3, 4*, 5*
d. Manual Initiation	NA	R(b)	Q	1, 2, 3, 4*, 5*
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"#				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R(a)	1, 2, 3
b. Drywell Pressure-High	S	M	R(a)	1, 2, 3
c. ADS Timer	NA	M	Q	1, 2, 3
d. Reactor Vessel Water Level - Low, Level 3	S	M	R(a)	1, 2, 3
e. LPCS Pump Discharge Pressure-High	S	M	R(a)	1, 2, 3
f. LPCI Pump A Discharge Pressure-High	S	M	R(a)	1, 2, 3
g. Manual Initiation	NA	R(b)	NA	1, 2, 3
B. DIVISION 2 TRIP SYSTEM				
1. RHR B AND C (LPCI MODE)				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R(a)	1, 2, 3, 4*, 5*
b. Drywell Pressure - High	S	M	R(a)	1, 2, 3
c. LPCI Pump B Start Time Delay Relay	NA	M	Q	1, 2, 3, 4*, 5*
d. Manual Initiation	NA	R(b)	Q	1, 2, 3, 4*, 5*
e. Reactor Vessel Pressure-Low (Injection Permissive)	S	M	R(a)	1,2,3,4*,5*

13. (NPE-85/15)

13. (NPE-85/15)

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The reactor coolant system leakage shall be demonstrated to be within each of the above limits by:

a. Monitoring the drywell atmospheric particulate and gaseous radioactivity at least once per 4 hours,

b. Monitoring the drywell floor and equipment drain sump level and flow rate at least once per 4 hours,

c. Monitoring the drywell air coolers condensate flow rate at least once per 4 hours, and

d. Monitoring the reactor vessel head flange leak detection system at least once per 24 hours.

4.4.3.2.2 Each reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1 shall be demonstrated OPERABLE by leak testing pursuant to Specification 4.0.5 and verifying the leakage of each valve to be within the specified limit:

a. At least once per 18 months, and

b. Prior to returning the valve to service following maintenance, repair or replacement work on the valve which could affect its leakage rate.

~~In addition, until the LPCS system and the RHR system injection valve reactor coolant pressure-low permissive is modified during or before the first refueling outage, the LPCS system check valve 1E21-F006 and the RHR system check valves 1E12-F041 A, B, and C shall also be demonstrated OPERABLE by verifying leakage to be within its limit:~~

~~1. Whenever the unit has been in COLD SHUTDOWN or REFUELING, after the last valve disturbance prior to reactor coolant system temperature exceeding 200°F.~~

~~2. Within 24 hours following valve disturbance except when in COLD SHUTDOWN or REFUELING.~~

The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

4.4.3.2.3 The high/low pressure interface valves leakage pressure monitors shall be demonstrated OPERABLE with alarm and interlock setpoints per Table 3.4.3.2-2 and Table 3.4.3.2-3 by performance of a:

a. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

b. CHANNEL CALIBRATION at least once per 18 months.

13. (NPE-85/15)

TABLE 3.4.3.2-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NUMBER	SYSTEM
E21-F005	LPCS
E21-F006	
E22-F004	HPCS
E22-F005	
E12-F008	RHR
E12-F009	
E12-F023	
E12-F041 A, B, C	
E12-F042 A, B, C	
E12-F050 A, B	
E12-F053 A, B	
E12-F308	
E12-F394	
E51-F063	RCIC
E51-F064	
E51-F065	
~~E51-F066~~	
E51-F076	
E51-F013	

(NPE-85/06)

TABLE 3.4.3.2-2

REACTOR COOLANT SYSTEM INTERFACE VALVES PRESSURE MONITORS - ALARM

VALVE NUMBER	SYSTEM	ALARM SETPOINT (psig)
E21-F005 to E21-F006	LPCS	~~≤ 50~~ 575
E12-F008 to E12-F006A	RHR	≤ 183
E12-F008 to E12-F006B	RHR	≤ 183
E12-F041A to E12-F042A	RHR	~~≤ 50~~ 475
E12-F041B to E12-F042B	RHR	~~≤ 50~~ 475
E12-F041C to E12-F042C	RHR	~~≤ 50~~ 475

13. (NPE-85/15)

TABLE 3.4.3.2-3

REACTOR COOLANT SYSTEM INTERFACE VALVES
PRESSURE INTERLOCKS

VALVE NUMBER	SYSTEM	INTERLOCK SETPOINT (psig)
E12-F052 to E51-F064	RCIC	$\leq$ 465
E12-F041A to E12-F042A	RHR	~~$\leq$ 50~~ 475
E12-F041B to E12-F042B	RHR	~~$\leq$ 50~~ 475
E12-F041C to E12-F042C	RHR	~~$\leq$ 50~~ 475
E21-F005 to E21-F006	LPCS	~~$\leq$ 50~~ 575