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-F019-A	19(O)	1	20
Main Steam Line Drains	B21-F016-B	19(I)		20
RHR Heat Exchanger "A" to CTMT SPR Sparger INL	E12-F028A-A	20(I)	5	90
RHR Heat Exchanger "A" to CTMT Pool	E12-F037A-A	20(I)	3	74
RHR Heat Exchanger "B" to CTMT SPR Sparger INL	E12-F028B-B	21(I)	5	90
RHR Heat Exchanger "B" to 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~~ P72-F123-B	36(I)	6A	33
~~Plant Service~~ Water Return	~~P44-F069-A~~ P72-F122-A	36(O)	6A	33
~~Plant Service~~ Water Supply	~~P44-F053-A~~ P72-F121-A	37(O)	6A	33
Chilled Water Supply	P71-F150-(A)	38(O)	6A	12

Drywell Chilled

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

Drywell Chilled

SYSTEM AND VALVE NUMBER		PENETRATION NUMBER	VALVE GROUP[a]	MAXIMUM ISOLATION TIME (Seconds)
Containment (Continued)				
RWCU Pump Suction	G33-F252-A	87(I)	8	35
RWCU Pump Suction	G33-F004-A	87(0)	8	35
RWCU Pump Disch.	G33-F053-B	88(I)	8	35
RWCU Pump Disch.	G33-F054-A	88(0)	8	35
b. Drywell				
Instrument Air	P53-F007-B	335(0)	6A	7
~~Plant Service~~ Water Return	~~P44-F076-A~~ P72-F125-A	331(I)	6A	32
~~Plant Service~~ Water Return	~~P44-F077-B~~ P72-F126-B	331(0)	6A	32
~~Plant Service~~ Water Return	~~P44-F074-B~~ P72-F124-B	332(0)	6A	32
RWCU Pump Suction	G33-F250-A	337(I)	8	35
RWCU Pump Suction	G33-F251-B	337(0)	8	35
Combustible Gas Con.	E61-F003B-B	338(0)	5	84
Combustible Gas Con.	E61-F003A-A	339(0)	5	84
Combustible Gas Con.	E61-F005A-A	340(0)	5	84
Combustible Gas Con.	E61-F005B-B	340(0)	5	84
Combustible Gas Con.	E61-F007-(A)	341(0)	5	9
Combustible Gas Con.	E61-F020-(B)	341(0)	5	18
Drywell Air Purge Supply	M41-F015-(A)	345(I)	7	4
Drywell Air Purge Supply	M41-F013-(B)	345(0)	7	4
Drywell Air Purge Exhaust	M41-F016-(A)	347(I)	7	4
Drywell Air Purge Exhaust	M41-F017-(B)	347(0)	7	4
Equipment Drains	P45-F009-(A)	348(I)	6A	6
Equipment Drains	P45-F010-(B)	348(0)	6A	6
Floor Drains	P45-F003-(A)	349(I)	6A	6
Floor Drains	P45-F004-(B)	349(0)	6A	6
Service Air	P52-F195-B	363(0)	6A	16
Chemical Sump Disch.	P45-F096-A	364(I)	6A	9
Chemical Sump Disch.	P45-F097-B	364(0)	6A	9
RWCU to Heat Exch.	G33-F253-B	366(0)	8	35
Reactor Water Sample Line	B33-F019-B	465(I)	10	36
Reactor Water Sample Line	B33-F020-A	465(0)	10	36

Amendment No. —

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM AND VALVE NUMBER		PENETRATION NUMBER
Containment (Continued)		
RHR Pump "A" Test Line to Suppr. Pool	E12-F227	23(O)[(e)]
RHR Pump "A" Test Line to Suppr. Pool	E12-F262	23(O)[(e)]
RHR Pump "A" Test Line to Suppr. Pool	E12-F228	23(O)[(e)]
RHR "A" Test Line to Supp. Pool	E12-F290A-A	23(O)[(d)]
RHR Pump "A" Test Line to Suppr. Pool	E12-F338	23(O)[(c)]
RHR Pump "A" Test Line to Suppr. Pool	E12-F339	23(O)[(c)]
RHR Pump "A" Test Line to Suppr. Pool	E12-F260	23(O)[(e)]
RHR Pump "C" Test Line to Suppr. Pool	E12-F280	24(O)[(d)]
RHR Pump "C" Test Line to Suppr. Pool	E12-F281	24(O)[(d)]
HPCS Suction	E22-F014	25(O)[(d)]
HPCS Discharge	E22-F005-(C)	26(I)
HPCS Discharge	E22-F218	26(I)
HPCS Discharge	E22-F201	26(I)
HPCS Test Line	E22-F035	27(O)[(d)]
HPCS Test Line	E22-F302	27(O)[(e)]
HPCS Test Line	E22-F301	27(O)[(e)]
LPCS Pump Suction	E21-F031	30(O)[(d)]
LPCS Discharge	E21-F006-(A)	31(I)
LPCS Discharge	E21-F200	31(I)
LPCS Discharge	E21-F207	31(I)
LPCS Test Line	E21-F217	32(O)[(d)]
LPCS Test Line	E21-F218	32(O)[(d)]
CRD Pump Discharge	C11-F122	33(I)
DCW ~~PSW~~ Supply	P72-F165 ~~P44-F043~~	37(I)
Plant Chilled Water Supply	P71-F151	38(I)
Service Air Supply	P52-F122	41(I)

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TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM AND VALVE NUMBER		PENETRATION NUMBER
Drywell (Continued)		
CRD to Recirc. Pump A Seals	B33-F017A	326(O)
Instrument Air	P53-F008	335(I)
Standby Liquid Control	C41-F007	328(I)
Standby Liquid Control	C41-F006	328(O)
Cont. Cooling Water Supply	P42-F115	329(I)
~~Plant Service~~ Drywell Chilled Water Supply	~~P44-F075~~ P72-F147	332(I)
Condensate Flush Conn.	B33-F204	333(I)
Condensate Flush Conn.	B33-F205	333(O)
Combustible Gas Control	E61-F002A	339(O)
Combustible Gas Control	E61-F002B	338(O)
Combustible Gas Control	E61-F004A	340(O)
Combustible Gas Control	E61-F004B	340(O)
Upper Containment Pool Drain	G41-F265	342(O)
CRD to Recirc. Pump B Seals	B33-F013B	346(I)
CRD to Recirc. Pump B Seals	B33-F017B	346(O)
Service Air	P52-F196	363(I)
Cont. Leak Rate Test Inst.	M61-F021	438A(I)
Cont. Leak Rate Sys.	M61-F020	438A(O)
BLIND FLANGES		
Cont. Leak Rate Sys.	NA	40(I)(O)
Cont. Leak Rate Sys.	NA	82(I)(O)
Containment Leak Rate System	NA	343(I)(O)

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM AND VALVE NUMBER		PENETRATION NUMBER
Containment (Continued)		
CRD T/C	C11-F128	33(O)
Cont. Purge Supply T/C	M41-F042	34(O)
Cont. Purge Exhaust T/C	M41-F051	35(O)
DCW ~~PSW~~ Supply T/C	P72-F167 ~~P44-F333~~	37(O)
Plant Chilled Water T/C	P71-F232	38(O)
Plant Chilled Water T/C	P71-F246	39(O)
Ctmt. Leak Rate T/C	M61-F009	40(I)
Service Air T/C	P52-F258	41(O)
Inst. Air T/C	P53-F036	42(O)
RWCU T/C	G33-F070	43(O)
CCW Supply T/C	P42-F161	44(O)
CCW Return T/C	P42-F162	45(I)
Condensate Supply T/C	P11-F095	56(O)
FPC & CU To Upper Cont. Pool T/C	G41-F340	57(I)
Aux. Bldg. Flr. & Equip. Drain Tk. to Suppr. Pool T/C	P45-F275	60(O)
Aux. Bldg. Flr. & Equip. Drain Tk. to Suppr. Pool T/C	P45-F290	60(O)
Stby. Liquid Control Sys. Mix. Tk. T/C (future use)	C41-F152	61(O)
Combustible Gas Control T/C	E61-F017	65(O)
Purge Radiation Detector T/C	M41-F054	66(O)
RHR "B" Test Line T/C	E12-F321	67(O)[(c)]
RHR "B" Test Line T/C	E12-F351	67(O)[(c)]
RHR "B" Test Line T/C	E12-F331	67(O)[(c)]

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
52-1112-21	800	0.100	480 V RECEPTACLE
52-1112-22	5	0.100	MOV-STM TUNNEL COOLER INLET ~~(N1P44F105A-N)~~ (N1P72F150A-N)
52-1112-24	32	0.100	MOV CLEANUP LINE RECIRC LOOP A (Q1G33F100-N)
52-1112-27	24	0.100	RESIN TANK AGITATOR (N1G36D020-N)
52-1112-28	38	0.100	MOV RWCU HEAT EXCHANGER BYPASS (N1G33F104-N)
52-1112-31	38	0.100	MOV RWCU HEAT EXCHANGER BYPASS (N1G33F044-N)
52-1112-36	500	0.100	REAC. RECIRC. PUMP SPACE HEATER (TB1B33C001A)
52-1112-37	800	0.100	480 V RECEPTACLE
52-1112-41	6	0.100	REAC RECIRC SAMPLE PANEL ISOL MOV (N1B33F129)
52-1113-07	125	0.100	CNTMT FLOOR DRAIN SUMP PUMP (N1P45C019B-N)
52-1113-21	60	0.100	DRYWELL EQUIP DRAIN SUMP PUMP (N1P45C002B-N)
52-1113-30	28	0.100	MOV RWCU HX OUTL ISOL VLV (N1G33F254-N)

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
52-1251-13	800	0.100	CNTMT CLR FAN COIL UNIT FAN (N1M41B001C-N)
52-1251-15	32	0.100	MOV - RWCS HX INL ISOL VLV (N1G33F256-N)
52-1251-18	38	0.100	MOV - REGEN HEAT EXCHANGER BYPASS (Q1G33F107-N)
52-1251-19	38	0.100	MOV - RWCU DRAIN FLOW ORIFICE BYP (N1G33F031-N)
52-1251-20	320	0.100	CNTMT EQUIP DRAIN PUMP (N1P45C004B-N)
52-1251-22	32	0.100	MOV - RWCU TO FLT "S" ISOL VLV (N1G33F255-N)
52-1251-26	1200	0.100	LIGHTING XFMR 1X112 (N1R18S112-D)
52-1251-28	5	0.100	MOV - STM TUNNEL COOLER INLET ~~(N1P44F105B-N)~~ (N1P72F150B-N)
52-1252-23	60	0.100	DRYWELL FLOOR DRAIN SUMP PUMP (N1P45C001B-N)
52-1252-27	500	0.100	FUEL TRANSFER SYS MN CONSOLE (N1F11E015-MC)
52-1411-01	38	0.100	MOV - VESSEL HEAD VENTILATION (Q1B21F002-N)

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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
52-1511-54	24	0.100	Spare
52-1521-02	6	0.100	MOV COMBUSTIBLE GAS CONTROL SYS (Q1E61F003A-A)
52-1521-03	6	0.100	MOV COMBUSTIBLE GAS CONTROL SYS (Q1E61F005A-A)
52-1521-07	10	0.100	MOV - SUPPR. POOL MAKE-UP VALVE (Q1E30F002A-A)
52-1521-14	600	0.100	SLC SYSTEM PUMP (Q1C41C001A-A)
52-1521-15	5	0.100	STORAGE TANK OUTLET VALVE (Q1C41F001A-A)
52-1521-28	12.5	0.100	MOV - INST LINE ISOL VALVE (Q1M71F595-A)
52-1521-44	10	0.100	MOV - SUPPR POOL MAKE-UP VALVE (Q1E30F001A-A)
52-1531-24	12.5	0.100	MOV - DRYWELL COOLER ISOLATION ~~(Q1P44F076-A)~~ (Q1P72F125-A)
52-1531-25	8	0.100	MOV - REACTOR WATER SAMPLE (Q1B33F020-A)

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
52-1542-10	320	0.100	DRYWELL COOLER FAN COIL UNIT (N1M51B006A-A)
52-1542-14	5	0.100	MOV - DRYWELL COOLER INLET ~~(N1P44F055-A)~~ (N1P72F145-A)
52-1542-15	5	0.100	MOV - DRYWELL COOLER INLET ~~(N1P44F057-A)~~ (N1P72F116-A)
52-1542-16	5	0.100	MOV - DRYWELL COOLER INLET ~~(N1P44F059-A)~~ (N1P72F139-A)
52-1542-17	5	0.100	MOV - DRYWELL COOLER INLET ~~(N1P44F061-A)~~ (N1P72F111-A)
52-1542-18	5	0.100	MOV - DRYWELL COOLER INLET ~~(N1P44F063-A)~~ (N1P72F101-A)
52-1542-19	5	0.100	MOV - DRYWELL COOLER INLET ~~(N1P44F065-A)~~ (N1P72F134-A)
52-1542-21	800	0.100	SLCS OPERATING HEATER (N1C41D002)
52-1542-22	24	0.100	DRWL PURGE COMP AUX OIL PUMP (Q1E61C001A-A)
52-1542-23	500	0.100	REFUELING PLATFORM ASSY (Q1F15E003-A)
52-1542-26	175	0.100	DRYWELL RECIRC FAN (N1M51C001-A)

Amendment No. ——

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~~ DCW CTMT STM TNL CLR ISOL ~~(Q1P44F070-B)~~ (Q1P72F123-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~~ DCW STEAM TUNNEL CLR ISOL ~~(Q1P44F074-B)~~ (Q1P72F124-B)
52-1611-43	12.5	0.100	MOV DCW ~~PSW~~ STEAM TUNNEL CLR ISOL ~~(Q1P44F077-B)~~ (Q1P72F126-B)
52-1611-44	38	0.100	MOV - SERVICE AIR DRYWELL ISOLATION (Q1P52F195-B)

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
52-1642-10	320	0.100	DRYWELL COOLER FAN COIL UNIT (N1M51B006B-B)
52-1642-14	12.5	0.100	MOV - DRYWELL COOLER INLET ~~(N1P44F056-B)~~ (N1P72F146-B)
52-1642-15	12.5	0.100	MOV - DRYWELL COOLER INLET ~~(N1P44F058-B)~~ (N1P72F117-B)
52-1642-16	12.5	0.100	MOV - DRYWELL COOLER INLET ~~(N1P44F060-B)~~ (N1P72F140-B)
52-1642-17	12.5	0.100	MOV - DRYWELL COOLER INLET ~~(N1P44F062-B)~~ (N1P72F112-B)
52-1642-18	12.5	0.100	MOV - DRYWELL COOLER INLET ~~(N1P44F064-B)~~ (N1P72F102-B)
52-1642-19	12.5	0.100	MOV - DRYWELL COOLER INLET ~~(N1P44F066-B)~~ (N1P72F135-B)
52-1642-21	24	0.100	DRWL PURGE COMP AUX OIL PUMP (Q1E61C001B-B)
52-1642-29	175	0.100	DRWL RECIRC FAN (N1M51C002B)

TABLE 3.8.4.1-1 (Continued)

PRIMARY CONTAINMENT PENETRATION CONDUCTOR
OVERCURRENT PROTECTIVE DEVICES

(e) 208/120 VAC Circuit Breakers (Continued)
GE Type THQB

BREAKER NUMBER	TIME O.C. PICKUP (Amperes)	RESPONSE TIME (Seconds)	SYSTEM/COMPONENT AFFECTED
52-1P222-27	15	4.0	DRWL. COOLERS SERVICE WATER CONT. TRANSMITTER (TT - N044)
52-1P251-13	15	4.0	PUMP VALVE SOLENOID CONT. CKT. & TEMPERATURE FOR REACTOR WATER CLEAN UP SYS.
52-1P252-37	15*	4.0	CONTAINMENT EQUIP. HATCH (Q1M23Y007-1)
52-1P252-38	15*	4.0	CONTAINMENT EQUIP. HATCH (Q1M23Y007-2)
52-1P411-19	15	4.0	Drywell Chilled ~~PLANT SERVICE~~ WATER SYS. CONTROL VALVE INDICATION ~~(1P44ZLR001)~~ (1P72ZLR018)
52-1P412-22	15	4.0	MOTOR SPACE HEATER FOR REACTOR RECIRC. SYSTEM (N1B33D003B1-N)
52-1P412-23	20	4.0	UTILITY POWER FOR REMOTE SIGNAL CONDITIONING PANEL
52-1P412-24	15	4.0	MOTOR SPACE HEATER FOR REACTOR RECIRC. SYSTEM (N1B33D003B2-N)
52-1P412-25	20	4.0	UTILITY POWER FOR REMOTE SIGNAL CONDITIONING PANEL

Drywell CW System

TABLE 3.8.4.2-1 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	BYPASS DEVICE (CONTINUOUS) (ACCIDENT CONDITIONS) (MO)	SYSTEM(S) AFFECTED
~~Q1P44F053~~ Q1P72F121	Continuous	~~Plant SW System~~
~~Q1P44F069~~ Q1P72F122	Continuous	~~Plant SW System~~
~~Q1P44F076~~ Q1P72F125	Continuous	~~Plant SW System~~
~~Q1P44F070~~ Q1P72F123	Continuous	~~Plant SW System~~
~~Q1P44F074~~ Q1P72F124	Continuous	~~Plant SW System~~
~~Q1P44F077~~ Q1P72F126	Continuous	~~Plant SW System~~
Q1P44F042	Continuous	Plant SW System
Q1P44F054	Continuous	Plant SW System
Q1P44F067	Continuous	Plant SW System
Q1P45F096	Continuous	Floor & Eqmt. Drain System
Q1P45F097	Continuous	Floor & Eqmt. Drain System
Q1P52F195	Continuous	Service Air System
Q1P53F003	Continuous	Instrument Air System
Q1P53F007	Continuous	Instrument Air System
Q1T48F005	Continuous	SGTS
Q1T48F006	Continuous	SGTS
Q1T48F024	Continuous	SGTS
Q1T48F026	Continuous	SGTS
Q1T48F023	Continuous	SGTS
Q1T48F025	Continuous	SGTS
Q1P45F273	Continuous	Floor & Eqmt. Drain System
Q1P45F274	Continuous	Floor & Eqmt. Drain System

2. (NLS-85/07)

SUBJECT:

Technical Specification 6.10.2.i, page 6-20

DISCUSSION:

The subject technical specification requires retention of "Records of Quality Assurance activities required by the Operational Quality Assurance Manual" for the duration of the Unit Operating License. It is proposed to add the words "not listed in Section 6.10.1" to the subject specification.

JUSTIFICATION:

The items specified in Technical Specifications 6.10.1.a, b and d are "Records of Quality Assurance activities required by the Operational Quality Assurance Manual." Specification 6.10.1 requires a retention period of five (5) years for these items, while 6.10.2 requires that such records be retained for the duration of the Unit Operating License. Appendix A to ANSI N45.2.9-1974 requires a retention period of five (5) years for the items listed in specifications 6.10.1.a, b & d, which indicates that 6.10.1 is the applicable specification for these items. The proposed change will clarify the technical specifications and preclude the occurrence of misinterpretation or conflicting requirements for these items.

SIGNIFICANT HAZARDS CONSIDERATION:

The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated because the change involves only a clarification to the technical specifications.

The proposed change does not create the possibility of a new or different kind of accident from any previously evaluated because the change is only a clarification and does not affect plant safety.

The proposed change does not involve a significant reduction in the margin of safety because the change is only administrative in nature and does not affect a margin of safety.

Therefore, the proposed change involves no significant hazards considerations.

6.10 RECORD RETENTION (Continued)

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those unit components identified in Table 5.7.1-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the unit staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the Operational Quality Assurance Manual not listed in Section 6.10.1

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the PSRC and the SRC.

l. Records of the service lives of all hydraulic and mechanical snubbers including the date at which the service life commences and associated installation and maintenance records.

m. Records of analyses required by the radiological environmental monitoring program.

3. (NPE-85/16)

SUBJECT: Technical Specifications 3.6.5 pages 3/4 6-46, -47.

DISCUSSION: Operating License Condition 2.C.(35), Post-LOCA Vacuum Breaker Position Indicators, states that prior to startup following the first refueling outage, Mississippi Power & Light (MP&L) shall install position indicators with redundant indication and alarm in the control room for the check valves associated with the Drywell Post-LOCA Vacuum Breakers.

MP&L is now in the process of developing a design change to satisfy the license condition. Technical specification changes are needed to supplement this design change. It is proposed to revise the subject technical specifications by deleting the temporary and 31 day surveillance requirements on the Drywell Post-LOCA Vacuum Breakers.

In support of the proposed technical specification changes, MP&L is providing as an attachment a description of the vacuum breaker position switch design for complying with O.L. condition 2.C.(35).

This design change is scheduled for implementation not later than startup following the first refueling outage. As done on several recent Technical Specification changes involving design changes to the plant, it is requested that the NRC issue the change with an open effective date and require that MP&L notify the NRC within 30 days of the effective date of implementation of the affected technical specification changes.

JUSTIFICATION: Amendment No. 4 to Facility Operating License NPF-13, dated October 14, 1982, added the Operating License condition and technical specification changes associated with this proposed change. The Safety Evaluation associated with Amendment 4 stated that MP&L intended to provide non-contact type position indication switches on the vacuum breaker check valves at a future date.

On May 24, 1985, in a letter to Mr. Harold R. Denton from Mr. O. D. Kingsley, MP&L described proposed changes to add single proximity type indicator switches (see attached drawing) to the vacuum breaker check valves. Separate position indication for each of the six check valves would be provided in the control room with a single common

annunciator alarm. The redundancy of position indication as required by the license condition would be met (1) between the vacuum relief lines by providing single switches on redundant check valves (i.e., F004 A and F004 B) and (2) within the vacuum relief lines by providing check valve and MOV isolation valve position indication (i.e., F004 A and F005 A) in each line.

In a response from the Nuclear Regulatory Commission (NRC), dated July 23, 1985, to the above submittal the NRC staff concluded that the design of the position switches for the drywell vacuum breaker check valves was acceptable.

The proposed change will delete the surveillance requirements placed in the Unit 1 technical specifications by Amendment No. 4 to NPF-13. The proposed change will also delete the 31 day surveillance requirements for the Drywell Post-LOCA vacuum breakers. Section XI of the ASME Boiler and Pressure Vessel Code for inservice pump and valve test procedures provides the rules and requirements for testing to verify operational readiness of valves (and their actuating and position indicating systems) in light-water cooled nuclear power plants. Section XI of the ASME code requires that the valves associated with the Post-LOCA Vacuum Breaker System be exercised at least once every 3 months unless such operation is not practical during plant operation. MP&L has been unable to determine the bases for the present 31 day surveillance interval in the technical specifications, however, MP&L believes that the 3 month test program as stated in Section XI of the ASME code is adequate to ensure safe and reliable valve operation during a postulated LOCA. In order to eliminate the need to revise plant procedures which reference the surveillance requirements of 4.6.5.b, MP&L proposes that the remaining requirements of 4.6.5.b not be renumbered, but that 4.6.5.b.1 be "DELETED."

Action requirements for the post-LOCA vacuum breaker motor operated isolation valves are included in technical specification 3.6.4 for the drywell isolation function. Present technical specification 3.6.5 contains the surveillance requirements in relation to the drywell post-LOCA vacuum breaker function but does not contain action requirements for these valves. In order to clarify the action statements, the proposed change will add the appropriate actions to be taken if an isolation valve is found to be inoperable. If an isolation valve is inoperable, the action statement will require the associated drywell post-LOCA vacuum breaker to be declared inoperable and that the provisions of Action a be followed for an inoperable post-LOCA vacuum breaker.

SIGNIFICANT HAZARDS CONSIDERATIONS:

The proposed amendment to the technical specifications results from a design change required to comply with Operating License Condition 2.C.(35). This design change adds position indicators on six check valves used as vacuum relief paths from the containment into the drywell. In addition to position indication, an alarm will be installed in the control room to indicate position movement of the six vacuum breaker check valves to the operators. This design change is conservative in nature in that it adds instrumentation not presently installed in the plant that will aid in ensuring that the vacuum breaker check valves stay in the closed position during normal plant operation.

The present surveillance requirements for the drywell post-LOCA vacuum breakers in specification 3/4.6.5 are augmented by a "Note 1" which applies until restart from the first refueling outage. The "Note 1" requirements were implemented due to the lack of remote position indication on the vacuum breakers, and are no longer needed when the above described design change is implemented. Consistent with commission guidance on applying the three criteria of 10CFR50.92(c), this proposed change would involve no significant hazards consideration. [48 FED. REG. 14864, 14870 (1983)]

The present 31 day surveillance requirement in specification 3/4.6.5 is not reflective of ASME Section XI testing intervals. Operability of the vacuum breakers including position indication is assured by compliance with ASME Section XI testing as implemented in Grand Gulf Surveillance procedures.

The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated because the addition of position indication and control room alarm for the drywell vacuum breakers is a required improvement to the present design and will aid operators in determining change of position of these valves. The change to the surveillance interval is consistent with ASME Section XI recommendations and will ensure operational readiness of the vacuum breakers.

The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated because the addition of position indication is a required improvement to present design and does not affect the accident analysis.

The proposed change does not involve a significant reduction in the margin of safety because no margin of safety is affected by this change. The proposed position indication will help assure proper positioning of the vacuum breakers and the proposed surveillances are consistent with ASME Section XI recommendations.

Therefore, the proposed change involves no significant hazards considerations.

3/4.6.5 DRYWELL POST-LOCA VACUUM BREAKERS

LIMITING CONDITION FOR OPERATION

3.6.5 All drywell post-LOCA vacuum breakers shall be OPERABLE and closed.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one drywell post-LOCA vacuum breaker inoperable for opening but known to be closed, restore the inoperable vacuum breaker to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With one drywell post-LOCA vacuum breaker open, restore the open vacuum breaker to the closed position within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With the position indicator of an OPERABLE drywell post-LOCA vacuum breaker inoperable, verify the vacuum breaker to be closed at least once per 24 hours by local indication. Otherwise declare the vacuum breaker inoperable. ~~(See Note 1)~~

d. With a vacuum breaker isolation valve inoperable, declare the associated drywell post-LOCA vacuum breaker inoperable and follow the requirements of ACTION a above.

SURVEILLANCE REQUIREMENTS

4.6.5 Each drywell post-LOCA vacuum breaker shall be:

a. Verified closed at least once per 7 days.

b. Demonstrated OPERABLE:

1. ~~At least once per 31 days by:~~
   ~~a) Cycling the vacuum breaker and isolation valve(s) through at least one complete cycle of full travel.~~
   ~~b) Verifying the position indicator OPERABLE by observing expected valve movement during the cycling test. (See Note 1)~~

Deleted

2. At least once per 18 months by:
   a) Verifying the pressure differential required to open the vacuum breaker, from the closed position, to be less than or equal to 1.0 psid, and ~~(See Note 1)~~
   b) Verifying the position indicator OPERABLE by performance of a CHANNEL CALIBRATION. ~~(See Note 1)~~

SURVEILLANCE REQUIREMENTS (Continued)

3. By verifying the OPERABILITY of the vacuum breaker isolation valve differential pressure actuation instrumentation with the opening setpoint of -1.0 to 0.0 psid (Drywell minus Containment) by performance of a:

   a) CHANNEL CHECK at least once per 24 hours,

   b) CHANNEL FUNCTIONAL TEST at least once per 31 days, and

   c) CHANNEL CALIBRATION at least once per 18 months.

~~Note 1: Until restart after the first refueling outage, the following requirements shall apply:~~

~~3.6.5~~

~~c. With the position indicator of an OPERABLE drywell post-LOCA isolation valve for a vacuum breaker inoperable, verify the isolation valve to be closed at least once per 24 hours by local indication. Otherwise declare the isolation valve inoperable.~~

~~4.6.5.b.1~~

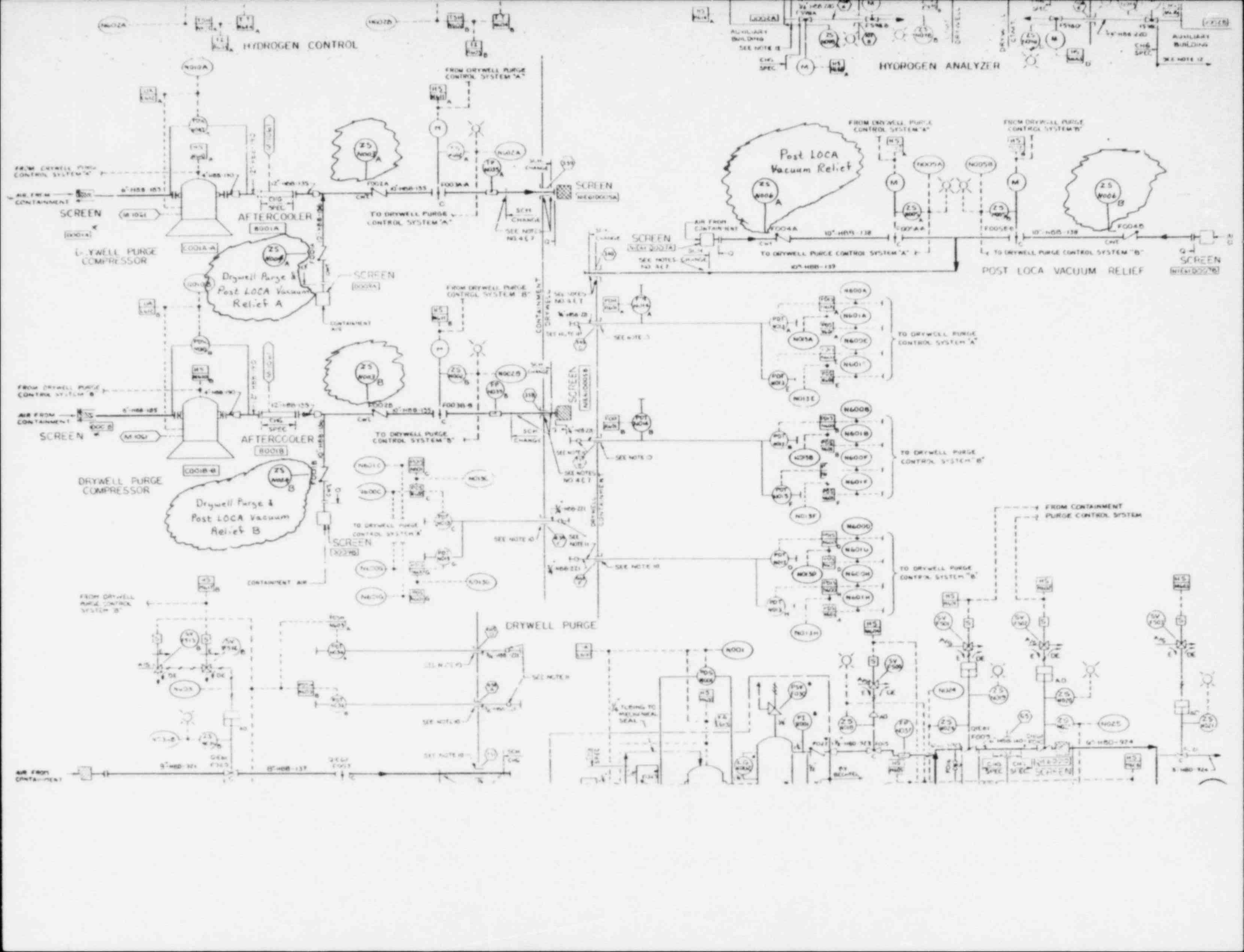
~~b. Verifying the position indicator for the vacuum breaker isolation valve OPERABLE by observing expected valve movement during the cycling test.~~

~~4.6.5.b.2~~

~~At least once per 18 months by:~~

~~a) Verifying the pressure differential required to open the vacuum breaker, from the closed position, to be less than or equal to 1.0 psid, and~~

~~b) Verifying the position indicator for the vacuum breaker isolation valve OPERABLE by performance of a CHANNEL CALIBRATION.~~



HYDROGEN CONTROL
HYDROGEN ANALYZER
SCREEN
AFTERCOOLER
DRYWELL PURGE COMPRESSOR
Drywell Purge & Post LOCA Vacuum Relief A
Drywell Purge & Post LOCA Vacuum Relief B
Post LOCA Vacuum Relief
POST LOCA VACUUM RELIEF
TO DRYWELL PURGE CONTROL SYSTEM "A"
TO DRYWELL PURGE CONTROL SYSTEM "B"
FROM DRYWELL PURGE CONTROL SYSTEM "A"
FROM DRYWELL PURGE CONTROL SYSTEM "B"
CONTAINMENT
DRYWELL
DRYWELL PURGE
FROM CONTAINMENT PURGE CONTROL SYSTEM
AIR FROM CONTAINMENT
CONTAINMENT AIR

4. (NS-85/02)

SUBJECT: Technical Specification 3.12.1.b, page 3/4 12-1

DISCUSSION: It is proposed to delete the reference to 6.9.1.13.f in the subject technical specification for the Radiological Environmental Monitoring Program. Mississippi Power & Light Company (MP&L) submitted a proposed change to Facility Operating License NPF-13 in a letter to Mr. Harold R. Denton from Mr. J. P. McGaughy (AECM-84/0319), dated June 22, 1984, deleting Section 6.9.1.13.f on thirty day written reports. The deletion of the reference to 6.9.1.13.f in the subject technical specification was inadvertently omitted in the June 22, 1984 submittal.

JUSTIFICATION: Section 6.9.1.13 of the Grand Gulf technical specifications concerning thirty day written reports was deleted in response to Generic Letter 83-43 issued by the NRC on December 19, 1983. This generic letter provided policy guidance regarding the implementation of technical specification changes required as a result of Section 50.73 to 10 CFR 50, "Licensee Event Reporting System." By deleting Section 6.9.1.13, MP&L intended to delete all references to that section from affected technical specifications. However, the deletion of the reference to 6.9.1.13 in technical specification 3.12.1.b was inadvertently missed and should be deleted at this time to provide an administratively correct specification.

SIGNIFICANT HAZARD CONSIDERATION:

The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated because this change is purely administrative in nature and reflects that section 6.9.1.13.f was deleted when Amendment 13 to Facility Operating License NPF-13 was issued.

The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated because no accident analyses are affected by this change.

The proposed change does not involve a significant reduction in the margin of safety because of the purely administrative nature of the change.

In its guidance on applying the criteria of 10CFR50.92(c), the Commission stated that amendments involving purely administrative changes, such as to correct an error or to achieve consistency throughout the technical specifications, would not involve significant hazards consideration. [48 FED. REG. 14864, 14870 (1983)] The proposed change falls into this category and therefore involves no significant hazards consideration.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.12.1-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12.1-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report per Specification 6.9.1.7, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity as the result of plant effluent in an environmental sampling medium at a specified location exceeding the reporting levels of Table 3.12.1-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose to a MEMBER OF THE PUBLIC is less than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2 and 3.11.2.3. ~~pursuant to Specification 6.9.1.13.f.~~ When more than one of the radionuclides in Table 3.12.1-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.12.1-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to a MEMBER OF THE PUBLIC is equal to or greater than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2 and 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

c. If milk or broad leaf vegetation sampling is relocated from one or more of the sample locations required by Table 3.12.1-1, identify new locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. In addition, report the cause(s) of the unavailability of samples and the new locations for obtaining replacement samples in the next Semi-annual Radioactive Effluent Release Report. Include in this report the revised ODCM figure(s) and table(s) reflecting the new locations. The specific locations from which samples were unavailable may then be deleted from the radiological environmental monitoring program and the table(s) in the ODCM, provided the locations from which the replacement samples were obtained are added to the table(s) as replacement locations.

d. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

5. (NPE-85/13)

SUBJECT: Technical Specification 3.1.3.1 Action d, Tables 3.3.1-1, 4.3.1.1-1 and 2.2.1-1, pages 3/4 1-4, 3/4 3-3, 3/4 3-8, and 2-4. Operating License page 14.

DISCUSSION: The proposed technical specification and operating license changes result from a design change to add diverse and redundant Scram Discharge Volume (SDV) level instrumentation and redundant vent and drain valves.

It is proposed to revise Tables 3.3.1-1 and 4.3.1.1-1 such that the applicable technical specification requirements for the SDV Water Level-High instrumentation apply to the transmitter/trip unit and the float switch separately. It is also proposed to revise Table 2.2.1-1 adding a list of both types of instrumentation to identify individual setpoint requirements.

Operating License Condition 2.C.(41) is proposed to be added to Facility Operating License NPF-29. This new license condition will provide a one time exception to the provisions of Specification 4.0.4 to allow entry into Operational Conditions 1 and 2 prior to performing Surveillance Requirement 4.1.3.1.4.a for the new scram discharge volume vent and drain valves.

This design change is scheduled for implementation not later than startup following the first refueling outage. As done on several recent Technical Specification changes involving design changes to the plant, it is requested that the NRC issue the change with an open effective date and require that MP&L notify the NRC within 30 days of the effective date of implementation of the affected technical specification changes.

JUSTIFICATION: The proposed design change associated with this technical specification change modifies the SDV design to meet the requirements of the NRC Generic study, "BWR Scram Discharge Volume System Safety Evaluation," of December 1, 1980. This design change has been required by the NRC for implementation prior to startup following the first scheduled refueling outage [Operating License Condition 2.C(15)]. The supporting technical specification change is conservative in nature since it adds requirements not previously incorporated into the technical specifications. The proposed operating license condition is necessary to allow testing of the new scram discharge volume vent and drain valves.

The purpose of this design change is to provide diverse redundant level switches (1C11-LSN013A-D) and redundant vent and drain valves (1C11-F180 and 1C11-F181) for scram discharge volume isolation. The level switches (1C11-LSN013A-D) are float type switches manufactured by Magnetrol (Model No. 5.0-751) and have an accuracy of $\pm$ .25 inches. These new switches provide independent trip signals to the reactor protection system in addition to the existing analog level transmitters (1C11-LT-N012A-D). Attached to this change is a drawing of the now existing design and a drawing showing placement of the new system components. This modification will minimize the potential for a common mode failure due to a drain or a vent valve not closing and resulting in an uncontrolled loss of coolant. This change affects the Reactor Protection System Instrumentation only. The Control Rod Block Instrumentation remains unchanged.

Technical Specification 3/4.1.3.1 presently provides Action Statement and Surveillance Requirements for scram discharge volume drain and vent valves. These present requirements are applicable to the new vent and drain valves being added by this design change. Present Action d is written to address only one scram discharge volume vent valve and one drain valve. The change to Action d will ensure that the present valves and the added vent and drain valves are equally addressed in the action provisions.

Entry into Operational Conditions 1 or 2 requires the 18 month surveillance requirement be completed on the scram discharge volume as stated in Specification 4.1.3.1.4.a. However, this specification requires a normal control rod configuration of less than or equal to 50% rod density which cannot be achieved without first entering Operational Condition 1 or 2. An exception to the provisions of Specification 4.0.4 for Surveillance Requirement 4.1.3.1.4.a was granted for the present scram discharge volume vent and drain valves by Amendment 10 to NPF-13 dated September 23, 1983. This previous one time exemption required Surveillance Requirement 4.1.3.1.4.a to be performed within 72 hours after achieving a normal control rod configuration of less than or equal 50% rod density. MP&L feels that this restriction requiring a reactor scram within 72 hours of reaching less than or equal to 50% rod density should not be imposed for the newly added scram discharge volume vent and drain valves. The new valves are in series with and located downstream of the present valves. The present valves provide an operable scram discharge volume and their operability is not affected by the addition of the new valves. Proposed License Condition 2.C.(41) will allow these new drain and vent valves to be tested per Specification 4.1.3.4.a during the first orderly shutdown but no later than the second refueling outage after reaching less than or equal to 50% rod density. Functional testing of the new valves will be accomplished as part of the design change package closure and prior to restart after the first refueling outage. This functional testing and the

performance of Specification 4.1.3.4.a as proposed by new License Condition 2.C.(41) will provide adequate assurance that the new vent and drain valves will perform properly as backups to the presently installed vent and drain valves.

Attached to this submittal is an update to the logic diagrams for the reactor protection system instrumentation in Table 3.3.1-1. These logics were originally submitted to the Nuclear Regulatory Commission (NRC) in a letter to Mr. Harold R. Denton from Larry F. Dale dated May 8, 1984 (AECM-84/0093).

SIGNIFICANT HAZARDS CONSIDERATION:

The design change associated with this proposed technical specification change will provide redundant trip signals to the reactor protection system when the scram discharge volume is filled with water. This redundant signal to RPS will help to ensure that the reactor is shutdown before the scram discharge volume is filled to the point that sufficient volume is not available to accept the discharge from the control rod drive. The addition of the redundant scram discharge volume vent and drain valves in series with the present valves provides additional assurance that the scram discharge volume will isolate on a reactor scram signal.

The proposed license condition allows a one time exception to the provisions of Specification 4.0.4 so that plant conditions necessary to satisfy the intent of Specification 4.1.3.1.4.a can be achieved. The requested exception to Specification 4.0.4 for the new vent and drain valves is similar to the exception granted by Amendment 10 to NPF-13 for the presently installed valves. The new vent and drain valves are backups to the present valves. The present valves are operable, perform the isolation function for the scram discharge volume, and do not require an exception to the provisions of Specification 4.0.4.

The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated because it is a required improvement that is conservative in nature since it adds requirements not currently listed in the technical specifications. The proposed license condition allows Surveillance Requirement 4.1.3.1.4.a to be performed for the new vent and drain valves at a scheduled plant shutdown thus preventing an unnecessary reactor scram. The present vent and drain valves are operable thus ensuring that the isolation function is performed when required. The new vent and drain valves are required improvements added as backups to the present valves. This design change increases the reliability of the reactor protection system and the scram discharge volume isolation function.

The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated because the design change associated with this technical specification change adds redundant reactor protection trip signals and redundant scram discharge volume isolation valves which increases the reliability of these systems.

The proposed change does not involve a significant reduction in the margin of safety because with the addition of redundant level switches and vent and drain valves, the margin of safety is increased. The probability of failures involving loss of coolant from the scram discharge volume is decreased with the addition of redundant vent and drain valves. The probability of receiving a reactor scram signal from high water level in the scram discharge volume is increased with the addition of the redundant level switches. Thus the reliability of the reactor protection system is increased by this design change.

Therefore, the proposed change involves no significant hazards considerations.

(b) Provide the second level undervoltage protection for Division 3 power supply (Item No. 373, T.S. Table 3.3.3-2).

(c) Incorporate a bypass or coincident logic in all Division 1 and 2 diesel generator protective trips, except for trips on diesel engine overspeed and generator differential current (Item No. 808, T.S. 4.8.1.1.2.d.16.d).

(38) Control Room Leak Rate (Section 6.2.6, SSER #6)

MP&L shall operate Grand Gulf Unit 1 with an allowable control room leak rate not to exceed 590 cfm. Upon restart of construction of Unit 2 control room, MP&L will be permitted to operate at a leak rate of 760 cfm as evaluated in SSER No. 6.

(39) Temporary Secondary Containment Boundary Change

For a period of time not to exceed 144 cumulative hours, the provisions of Specification 3/4.6.6.1 may be applied to the railroad bay area including the exterior railroad bay door on the auxiliary building in lieu of the present secondary containment boundaries that isolate the railroad bay area. While the railroad bay area is being used as a secondary containment boundary, the railroad bay door may be opened for the purpose of moving trucks in and out provided the four hour limitation in ACTION a of Technical Specification 3.6.6.1 is reduced to one hour. A fire watch shall be established in the railroad bay area while the door is being used as a secondary containment boundary.

(40) Temporary Ultimate Heat Sink Change

With the plant in OPERATIONAL condition 4, SSW cooling tower basin A may be considered OPERABLE in accordance with Technical Specification 3.7.1.3 with less than a 30 day supply of water (without makeup) during the time that SSW basin B is drained to replace its associated service water pump provided:

(a) SSW basin A water level is maintained greater than or equal to 87".

(b) At least two sources of water (other than normal makeup with one source not dependent on offsite power) are available for makeup to SSW basin A.

This license condition may remain in effect until plant startup following the outage scheduled for fall 1985.

(41) See attached change.

Insert to Operating License Page 14

(41) Scram Discharge Volume Test

For the scram discharge volume surveillance test required in Section 4.1.3.1.4.a, the provisions of Specification 4.0.4 are suspended provided that the surveillance requirement is performed during the first orderly shutdown but no later than the second refueling outage after reaching less than or equal to 50% rod density in OPERATIONAL CONDITIONS 1 or 2. This exception applies to the scram discharge volume vent and drain valves added by design change to comply with License Condition 2.C.(15).

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

2. If the inoperable control rod(s) is inserted, within one hour disarm the associated directional control valves** either:

   a) Electrically, or

   b) Hydraulically by closing the drive water and exhaust water isolation valves.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

3. The provisions of Specification 3.0.4 are not applicable.

c. With more than 8 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

d. With a ~~the~~ scram discharge volume vent valve and/or a ~~the~~ scram discharge volume drain valve inoperable, close the inoperable valve(s) within 1 hour, restore the inoperable valve(s) to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The scram discharge volume drain and vent valves shall be demonstrated OPERABLE by:

a. At least once per 31 days verifying each valve to be open,* and

b. At least once per 92 days cycling each valve through at least one complete cycle of full travel.

4.1.3.1.2 When above the low power setpoint of the RPCS, all withdrawn control rods not required to have their directional control valves disarmed electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch:

a. At least once per 7 days, and

b. At least once per 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference.

4.1.3.1.3 All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.1.3.2, 4.1.3.3, 4.1.3.4 and 4.1.3.5.

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*These valves may be closed intermittently for testing under administrative controls.

**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
9. Scram Discharge Volume Water Level - High	~~1, 2~~ ~~5(g)~~	~~2~~ ~~2~~	~~1~~ ~~3~~
a. Transmitter/Trip Unit	1, 2 5(g)	2 2	1 3
b. Float Switch	1, 2 5(g)	2 2	1 3
10. Turbine Stop Valve - Closure	1(h)	4	6
11. Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure - Low	1(h)	2	6
12. Reactor Mode Switch Shutdown Position	1, 2 3, 4 5	2 2 2	1 7 3
13. Manual Scram	1, 2 3, 4 5	2 2 2	1 8 9

a. Transmitter/Trip Unit	S	M	R[g]	1,2,5[l]
b. Float Switch	NA	M	R	1,2,5[l]

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
9. Scram Discharge Volume Water Level - High	~~S~~	~~M~~	~~R[g]~~	~~1, 2, 5[l]~~
10. Turbine Stop Valve - Closure	S	M	R[g]	1
11. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	S	M	R[g]	1
12. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
13. Manual Scram	NA	M	NA	1, 2, 3, 4, 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 decade during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least 1/2 decade during each controlled shutdown, if not performed within the previous 7 days.
(c) [DELETED]
(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.
(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 MWD/T using the TIP system.
(g) Calibrate trip unit at least once per 31 days.
(h) Verify measured drive flow to be less than or equal to established drive flow at the existing flow control valve position.
(i) This calibration shall consist of verifying the 6 ± 1 second simulated thermal power time constant.
(j) Not applicable when the reactor pressure vessel head is unbolted or removed per Specification 3.10.1.
(k) Not applicable when DRYWELL INTEGRITY is not required.
(l) Applicable with any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 2.2.1-1

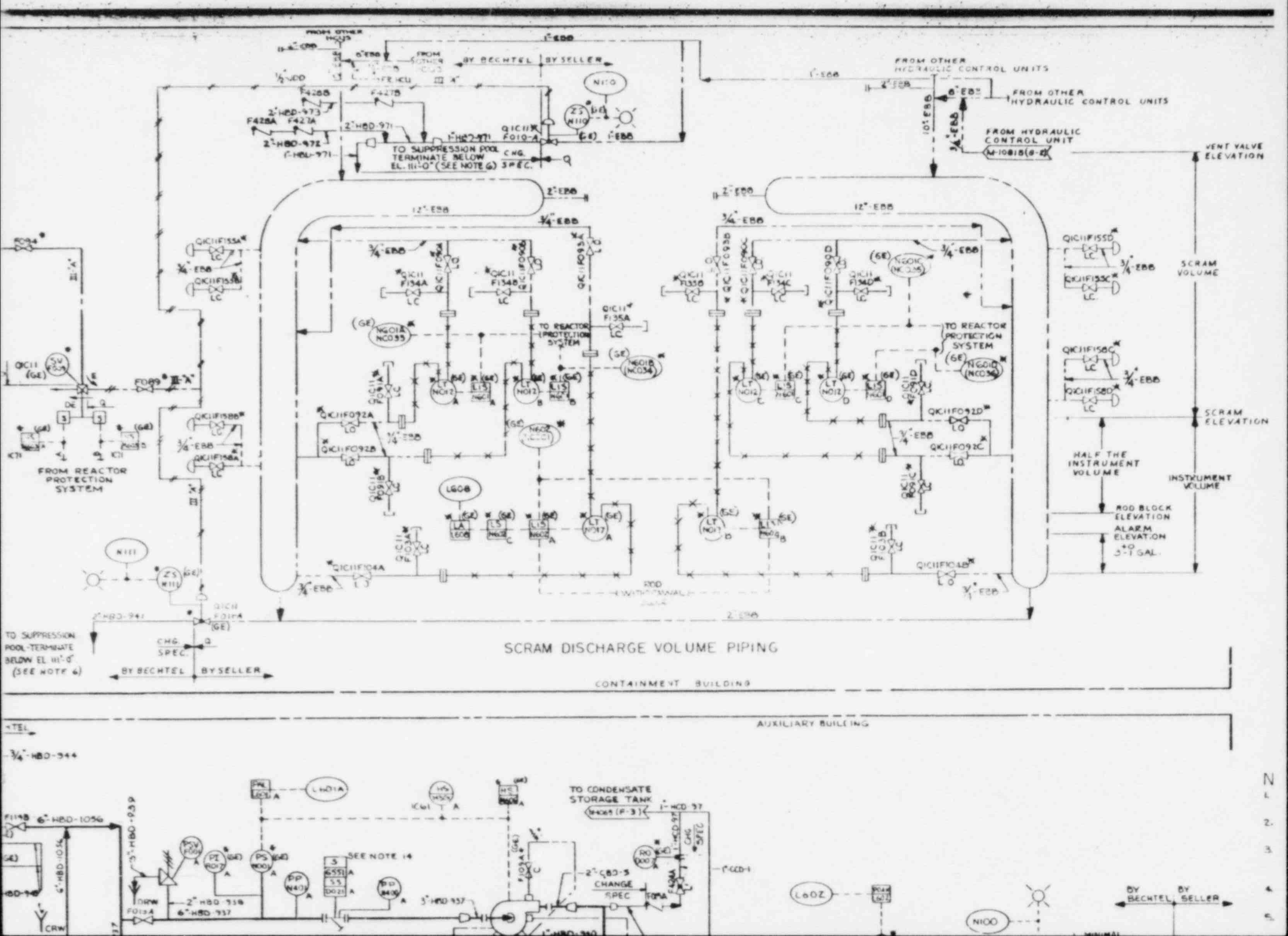
REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	≤ 120/125 divisions of full scale	≤ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-High, Setdown	≤ 15% of RATED THERMAL POWER	≤ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power-High		
1) Flow Biased	≤ 0.66 W+48%, with a maximum of	≤ 0.66 W+51%, with a maximum of
2) High Flow Clamped	≤ 111.0% of RATED THERMAL POWER	≤ 113.0% of RATED THERMAL POWER
c. Neutron Flux-High	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
d. Inoperative	NA	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ 1064.7 psig	≤ 1079.7 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 11.4 inches above instrument zero*	≥ 10.8 inches above instrument zero*
5. Reactor Vessel Water Level-High, Level 8	≤ 53.5 inches above instrument zero*	≤ 54.1 inches above instrument zero*
6. Main Steam Line Isolation Valve - Closure	≤ 6% closed	≤ 7% closed
7. Main Steam Line Radiation - High	≤ 3.0 x full power background	≤ 3.6 x full power background
8. Drywell Pressure - High	≤ 1.23 psig	≤ 1.43 psig
9. Scram Discharge Volume Water Level - High	~~≤ 60% of full scale~~	~~≤ 63% of full scale~~
10. Turbine Stop Valve - Closure	≥ 40 psig**	≥ 37 psig
11. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥ 44.3 psig**	≥ 42 psig
12. Reactor Mode Switch Shutdown Position	NA	NA
13. Manual Scram	NA	NA

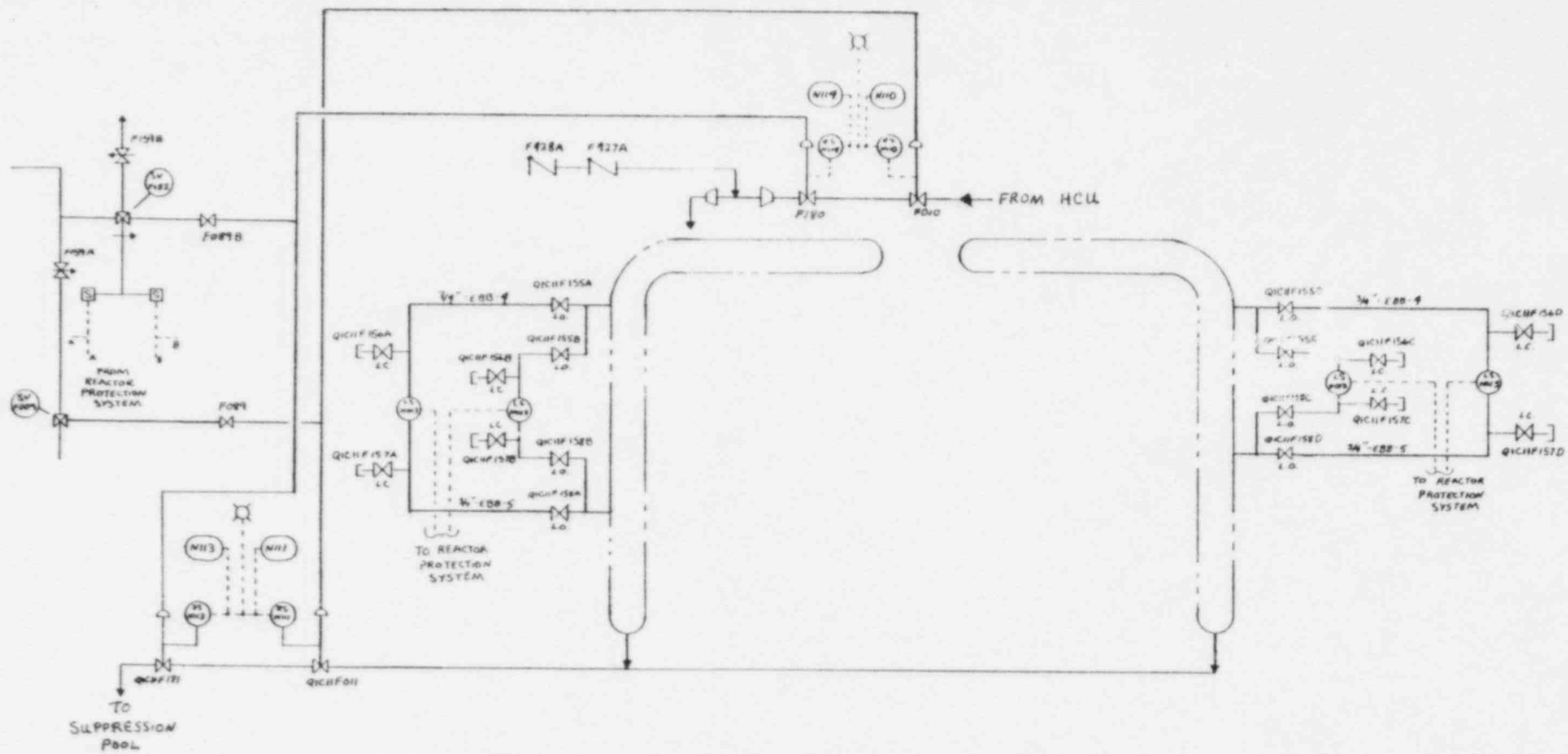
*See Bases Figure B 3/4 3-1.

**Initial setpoint. Final setpoint to be determined during startup test program. Any required change to this setpoint shall be submitted to the Commission within 90 days of test completion.

a. Transmitter/Trip Unit	≤ 60% of full scale	≤ 63% of full scale
b. Float Switch	≤ 64"	≤ 65"



FROM OTHER HYDRAULIC CONTROL UNITS
FROM OTHER HYDRAULIC CONTROL UNITS
FROM HYDRAULIC CONTROL UNIT
M-10818(B-8)
BY BECHTEL
BY SELLER
TO SUPPRESSION POOL TERMINATE BELOW EL. 111'-0" (SEE NOTE 6)
CHG. SPEC.
VENT VALVE ELEVATION
SCRAM VOLUME
SCRAM ELEVATION
HALF THE INSTRUMENT VOLUME
INSTRUMENT VOLUME
ROD BLOCK ELEVATION
ALARM ELEVATION
TO REACTOR PROTECTION SYSTEM
TO REACTOR PROTECTION SYSTEM
FROM REACTOR PROTECTION SYSTEM
ROD WITHDRAWAL
TO SUPPRESSION POOL-TERMINATE BELOW EL 111'-0" (SEE NOTE 6)
BY BECHTEL
BY SELLER
SCRAM DISCHARGE VOLUME PIPING
CONTAINMENT BUILDING
AUXILIARY BUILDING
TO CONDENSATE STORAGE TANK
SEE NOTE 14
CHANGE SPEC
BY BECHTEL
BY SELLER


FROM HCU
F128A
F127A
F140
F010
F089B
F089
FROM REACTOR PROTECTION SYSTEM
TO REACTOR PROTECTION SYSTEM
TO REACTOR PROTECTION SYSTEM
TO SUPPRESSION POOL
N113
N112
N119
N110
Q1C11F155A
Q1C11F156A
Q1C11F157A
Q1C11F011
Q1C11F157D
Q1C11F156D

DEFINITIONS FOR
"CHANNELS", "TRIP SYSTEMS", AND "TRIP FUNCTIONS"
FOR REACTOR PROTECTION SYSTEM INSTRUMENTATION TABLE 3.3.1-1

Trip Unit	Parameter	Logic
C51-K601A	(3)IRM-Neutron Flux High	
C51-K601E	(3)IRM-Neutron Flux High	
C51-K601A	(3)IRM-Inoperative	
C51-K601E	(3)IRM-Inoperative	
C51-Z405A	(3)APRM-Neutron Flux High, Setdown	
C51-Z405E	(3)APRM-Neutron Flux High, Setdown	
C51-Z409A	APRM-Flow Biased Thermal Power-Hi	
C51-Z409E	APRM-Flow Biased Thermal Power-Hi	
C51-Z401A	APRM-Neutron Flux High	
C51-Z401E	APRM-Neutron Flux High	
C51-Z401A	APRM-Inoperative	
C51-Z401E	APRM-Inoperative	
B21-PIS-N678A	*Rx Steam Dome Pressure High	Any One — "A" Trip Logic
B21-LIS-N680A	RPV Level 3	
B21-LS-N683A	(1)RPV Level 8	
B21-ZS-N101A	(1)MSIV Closure	Either
B21-ZS-N102A	(1)MSIV Closure	Both
B21-ZS-N101D	(1)MSIV Closure	
B21-ZS-N102D	(1)MSIV Closure	Either
D17-RITS-K610A	MSL Radiation High	
C71-PIS-N650A	Drywell Pressure High	
C11-LIS-N601A	SDV Water Level High	
C11-LS-N013A	SDV Water Level High	
C71-PIS-N606A	(2)Turb Stop Valve Closure	Both
C71-PIS-N606E	(2)Turb Stop Valve Closure	
C71-PIS-N605A	(2)Turb Control Valve Fast Closure	
C71-HSS-M602 **	Rx Mode Switch Shutdown	
C71-HS-M600A	Manual Scram	

(1) This CHANNEL is automatically bypassed when the Reactor Mode Switch is not in the RUN position.

(2) This CHANNEL is automatically bypassed when turbine first stage pressure is less than 30% of the value of turbine first stage pressure at valves wide open steam flow, equivalent to less than 40% rated thermal power.

(3) This CHANNEL is automatically bypassed when the Reactor Mode Switch is in the RUN position.

* One Channel (Typical of 28 shown on this page)

** There is only one Reactor Mode Switch which operates relays in both Trip Systems A and B. The Reactor Mode Switch and its associated contacts constitute one Channel.

DEFINITIONS FOR
"CHANNELS", "TRIP SYSTEMS", AND "TRIP FUNCTIONS"
FOR REACTOR PROTECTION SYSTEM INSTRUMENTATION TABLE 3.3.1-1 (Continued)

Trip Unit	Parameter	Logic
C51-K601C	(3)IRM-Neutron Flux High	
C51-K601G	(3)IRM-Neutron Flux High	
C51-K601C	(3)IRM-Inoperative	
C51-K601G	(3)IRM-Inoperative	
C51-Z405C	(3)APRM-Neutron Flux High, Setdown	
C51-Z405G	(3)APRM-Neutron Flux High, Setdown	
C51-Z409C	APRM-Flow Biased Thermal Power-Hi	
C51-Z409G	APRM-Flow Biased Thermal Power-Hi	
C51-Z401C	APRM-Neutron Flux High	
C51-Z401G	APRM-Neutron Flux High	
C51-Z401C	APRM-Inoperative	
C51-Z401G	APRM-Inoperative	
B21-PIS-N678C	*Rx Steam Dome Pressure High	Any One — "C" Trip Logic
B21-LIS-N680C	RPV Level 3	
B21-LS-N683C	(1)RPV Level 8	
B21-ZS-N101C	(1)MSIV Closure	Either
B21-ZS-N102C	(1)MSIV Closure	Both
B21-ZS-N101B	(1)MSIV Closure	
B21-ZS-N102B	(1)MSIV Closure	Either
D17-RITS-K610C	MSL Radiation High	
C71-PIS-N650C	Drywell Pressure High	
C11-LIS-N601C	SDV Water Level High	
C11-LS-N013C	SDV Water Level High	
C71-PIS-N606C	(2)Turb Stop Valve Closure	Both
C71-PIS-N606G	(2)Turb Stop Valve Closure	
C71-PIS-N605C	(2)Turb Control Valve Fast Closure	
C71-HSS-M602	**Rx Mode Switch Shutdown	
C71-HS-M600C	Manual Scram	

(1) This CHANNEL is automatically bypassed when the Reactor Mode Switch is not in the RUN position.

(2) This CHANNEL is automatically bypassed when turbine first stage pressure is less than 30% of the value of turbine first stage pressure at valves wide open steam flow, equivalent to less than 40% rated thermal power.

(3) This CHANNEL is automatically bypassed when the Reactor Mode Switch is in the RUN position.

* One Channel (Typical of 28 shown on this page)

** There is only one Reactor Mode Switch which operates relays in both Trip Systems A and B. The Reactor Mode Switch and its associated contacts constitute one Channel.

DEFINITIONS FOR
"CHANNELS", "TRIP SYSTEMS", AND "TRIP FUNCTIONS"
FOR REACTOR PROTECTION SYSTEM INSTRUMENTATION TABLE 3.3.1-1 (Continued)

Trip Unit	Parameter	Logic
C51-K601B	(3)IRM-Neutron Flux High	
C51-K601F	(3)IRM-Neutron Flux High	
C51-K601B	(3)IRM-Inoperative	
C51-K601F	(3)IRM-Inoperative	
C51-Z405B	(3)APRM-Neutron Flux High, Setdown	
C51-Z405F	(3)APRM-Neutron Flux High, Setdown	
C51-Z409B	APRM-Flow Biased Thermal Power-Hi	
C51-Z409F	APRM-Flow Biased Thermal Power-Hi	
C51-Z401B	APRM-Neutron Flux High	
C51-Z401F	APRM-Neutron Flux High	
C51-Z401B	APRM-Inoperative	
C51-Z401F	APRM-Inoperative	
B21-PIS-N678B	*Rx Steam dome pressure high	Any One — "B" Trip Logic
B21-LIS-N680B	RPV Level 3	
B21-LS-N683B	(1)RPV Level 8	
B21-ZS-N101A	(1)MSIV Closure	Either
B21-ZS-N102A	(1)MSIV Closure	Both
B21-ZS-N101B	(1)MSIV Closure	
B21-ZS-N102B	(1)MSIV Closure	Either
D17-RITS-K610B	MSL Radiation High	
C71-PIS-N650B	Drywell Pressure High	
C11-LIS-N601B	SDV Water Level High	
C11-LS-N013B	SDV Water Level High	
C71-PIS-N606B	(2)Turb Stop Valve Closure	Both
C71-PIS-N606F	(2)Turb Stop Valve Closure	
C71-PIS-N605B	(2)Turb Control Valve Fast Closure	
C71-HSS-M602 **	Rx Mode Switch Shutdown	
C71-HS-M600B	Manual Scram	

(1) This CHANNEL is automatically bypassed when the Reactor Mode Switch is not in the RUN position.

(2) This CHANNEL is automatically bypassed when turbine first stage pressure is less than 30% of the value of turbine first stage pressure at valves wide open steam flow, equivalent to less than 40% rated thermal power.

(3) This CHANNEL is automatically bypassed when the Reactor Mode Switch is in the RUN position.

* One Channel (Typical of 28 shown on this page)

** There is only one Reactor Mode Switch which operates relays in both trip systems A and B. The Reactor Mode Switch and its associated contacts constitute one Channel.

DEFINITIONS FOR
"CHANNELS", "TRIP SYSTEMS", AND "TRIP FUNCTIONS"
FOR REACTOR PROTECTION SYSTEM INSTRUMENTATION TABLE 3.3.1-1 (Continued)

Trip Unit	Parameter	Logic
C51-K601D	(3)IRM-Neutron Flux High	
C51-K601H	(3)IRM-Neutron Flux High	
C51-K601D	(3)IRM-Inoperative	
C51-K601H	(3)IRM-Inoperative	
C51-Z405D	(3)APRM-Neutron Flux High, Setdown	
C51-Z405H	(3)APRM-Neutron Flux High, Setdown	
C51-Z409D	APRM-Flow Biased Thermal Power-Hi	
C51-Z409H	APRM-Flow Biased Thermal Power-Hi	
C51-Z401D	APRM-Neutron Flux High	
C51-Z401H	APRM-Neutron Flux High	
C51-Z401D	APRM-Inoperative	
C51-Z401H	APRM-Inoperative	
B21-PIS-N678D	*Rx Steam Dome Pressure High	Any One — "D" Trip Logic
B21-LIS-N680D	RPV Level 3	
B21-LS-N683D	(1)RPV Level 8	
B21-ZS-N101D	(1)MSIV Closure	Either — Both
B21-ZS-N102D	(1)MSIV Closure	
B21-ZS-N101C	(1)MSIV Closure	
B21-ZS-N102C	(1)MSIV Closure	Either
D17-RITS-K610D	MSL Radiation High	
C71-PIS-N650D	Drywell Pressure High	
C11-LIS-N601D	SDV Water Level High	
C11-LS-N013D	SDV Water Level High	
C71-PIS-N606D	(2)Turb Stop Valve Closure	Both
C71-PIS-N606H	(2)Turb Stop Valve Closure	
C71-PIS-N605D	(2)Turb Control Valve Fast Closure	
C71-HSS-M602	**Rx Mode Switch Shutdown	
C71-HS-M600D	Manual Scram	

The TRIP FUNCTION for Reactor Protection System logics on pages 2 through 5 is shown below.

Trip Logic	Logic	Trip System	Logic	Trip Function
"A" Trip Logic "C" Trip Logic	Either One	De-energizes scram pilot solenoids in TRIP SYSTEM A	Both	REACTOR SCRAM
"B" Trip Logic "D" Trip Logic	Either One	De-energizes scram pilot solenoids in TRIP SYSTEM B		

(1) This CHANNEL is automatically bypassed when the Reactor Mode Switch is not in the RUN position.
(2) This CHANNEL is automatically bypassed when turbine first stage pressure is less than 30% of the value of turbine first stage pressure at valves wide open steam flow, equivalent to less than 40% rated thermal power.
(3) This CHANNEL is automatically bypassed when the Reactor Mode Switch is in the RUN position.
* One Channel (Typical of 28 shown on this page)
** There is only one Reactor Mode Switch which operates relays in both trip systems A and B. The Reactor Mode Switch and its associated contacts constitute one Channel.

6. (NLS-85/16)

SUBJECT: Technical Specification Tables 3.3.3-1 and 4.3.3.1-1, pages 3/4 3-29 and 3/4 3-36.

DESCRIPTION OF CHANGE: The proposed change modifies the ## note of the subject tables by deleting the phrase "Prior to STARTUP following the first refueling outage." The deletion of this phrase from the ## note to the two tables retains the requirements through the first refueling outage and extends the requirements to be applicable for subsequent plant operations. The notes modify the technical specifications on the High Pressure Core Spray (HPCS) system actuation instrumentation such that the injection function of Drywell Pressure-High and Manual Initiation are not required to be OPERABLE when the indicated water level on the wide range instrument is greater than Level 8 coincident with the reactor pressure being less than 600 psig.

DISCUSSION: The primary purpose of the HPCS is to maintain reactor vessel inventory in the event of small pipe breaks (1 inch nominal diameter or smaller) which do not depressurize the reactor vessel. HPCS also provides spray cooling heat transfer during breaks in which the core is calculated to uncover. If a loss-of-coolant accident should occur, a low reactor vessel water level signal or a high drywell pressure signal initiates the HPCS and its support equipment. The system can also be placed in operation manually.

With a high reactor vessel water level interlock (Level 8) actuated, the HPCS is automatically stopped by a signal to the injection valve to close. This prevents undesirable spillover of water into the main steamlines. HPCS operation is thus terminated until the low water level initiation setpoint is reached or the isolation logic is manually reset once the Level 8 interlock clears.

Reactor vessel water level is monitored by level transmitters that sense the difference between the pressure due to a constant reference leg of water and the pressure due to the actual height of water in the vessel. The instrumentation is the condensate chamber reference leg type (See Figure 1.) These instruments are strictly differential pressure devices which are reactor coolant density sensitive and are calibrated to be most accurate at the specific vessel conditions appropriate for the associated system functions actuated by the instrumentation.

The shutdown water level range and fuel zone water level range instruments are calibrated to read accurately at atmospheric pressure; the upset, narrow, and wide range water level instruments are calibrated for normal operating conditions (saturated steam at 1025 psig.) Figures 2 and 3 depict the level ranges and setpoints. At low coolant temperatures and pressures, those instruments calibrated for normal operating conditions will read higher than actual level.

The HPCS discharge valve is interlocked closed at the vessel Level 8 setpoint (setpoint less than or equal to 53.5" above instrument zero or 220" above the active fuel.) An artificially high level indication at low pressure may result in HPCS isolation when the actual vessel level is below the Level 8 setpoint. The isolation logic may be manually reset once the indicated vessel level drops below this setpoint or it will reset automatically when the indicated vessel level reaches the Level 2 HPCS initiation setpoint (a level of -41.6" below instrument zero or 125" above the fuel.)

Technical Specification Table 3.3.3-1. requires HPCS Drywell Pressure-High and Manual Initiation Actuation Instrumentation to be OPERABLE in various Operational Conditions. Actuation of these devices will result in vessel injection unless reactor vessel level is above the Level 8 setpoint or the Level 8 isolation has not been reset. The ## notes to Technical Specification Tables 3.3.3-1 and 4.3.3.1-1 state that the injection function of Drywell Pressure-High and Manual Initiation are not required to be OPERABLE at times when the indicated Level 8 isolation signal is present coincident with the reactor pressure at or below 600 psig.

Deletion of the phrase "Prior to STARTUP following the first refueling outage" modifies the ## notes such that the requirements remain applicable through the first refueling outage and are extended throughout subsequent plant operations. By imposing the requirements on subsequent plant operations, design modifications and concomitant potential problems are negated. The notes were initially added to Technical Specifications Tables 3.3.3-1 and 4.3.3.1-1 to address a concern with respect to the reactor vessel level instrumentation. The requirements were imposed through the first refueling outage which allowed adequate time to consider the necessity for a design modification. A design modification has been determined to be unnecessary because the requirements as currently imposed are considered adequate to ensure the continued safe operation of the plant. The following is a discussion of the reactor water level instrumentation and concern, which prompted the imposition of the requirements, to indicate that the concern has been properly addressed, a redesign is not required, and the subject change to the ## notes is justified.

JUSTIFICATION: MASS OF COOLANT: The wide range level instrumentation measures the mass of coolant above the lower instrument tap and displays this as level. Since the instrument is not compensated for changes in water density, the interlocks occur with the same mass of coolant in the vessel regardless of the pressure or actual water level. That is to say, mass is the variable that is being indicated by level and for a specific indicated level the mass is the same regardless of pressure. Because the mass of coolant is the same at the time of HPCS initiation, the system response to a loss of coolant event at low pressure is essentially the same as that previously analyzed under normal

operating pressure (1025 psig), and those analyses are applicable to the low pressure condition. Mass, not density nor indicated level, is the true measure of the amount of coolant available for an event.

ACCIDENT ANALYSIS: Only one ECCS accident analysis previously presented in the FSAR assumed HPCS initiation by a high drywell pressure signal - a steamline break inside the containment. This event was reanalyzed with the premise that the high drywell initiation feature is defeated and HPCS is initiated only by low water level (Level 2). For this event with the worst single failure (LPCS diesel generator failure), the peak cladding temperature (PCT) was calculated to be 1322°F, which is significantly below the criteria of 2200°F as given in 10 CFR 50.46. Thus, initiation of HPCS only by low water level, and not by a high drywell pressure signal, is not a significant safety concern. The results of the re-analysis are attached and include copies of Figures 6.3-8, and 6.3-63 through 6.3-66 from the recently updated FSAR.

LEVEL TRIP IMPACT: When the vessel is at low pressure, the Level 2 trip will occur approximately 1 foot below the allowable Level 2 limit. This is not a safety concern because the mass of coolant available for boil off during a loss of inventory event (either transient or LOCA) is the same as if the vessel was at normal operating conditions.

The wide range Level 1 trip for ECCS is unaffected because the density compensation error is negligible at this low water level.

With the vessel at low pressure, the level 8 interlock will occur with actual water level lower than with the reactor hot and at high pressure; thus, for the purpose of preventing vessel overfill, the actuation of the interlock will be in the conservative direction and there is no safety concern.

With the reactor cold and actual water level at the normal level (between Level 4 = 32.7" and Level 7 = 40.7"), the Level 8 interlock will occur due to the artificial high level indication. This is not a safety concern since no water addition is needed until the water level drops to Level 2. Further, with the actual level at Level 7, reset of the Level 8 interlock will occur at approximately 600 psig (MP&L letter to NRC dated September 13, 1983 - AECM-83/0557.)

LOW PRESSURE EFFECTS: With the vessel at low pressure, the plant is normally at reduced power. Also, at low pressure and saturated conditions, the energy required to boil water is greater than at high pressure and saturated conditions. Both of these effects reduce the boil-off rate of the coolant when compared to that of the safety analyses for high pressure.

For LOCA events, the coolant loss out of a break is less at low pressure conditions; and, low pressure ECCS injection occurs earlier as a result of the initial low pressure. Also, a number of systems other than HPCS would be more responsive to the mitigation of accidents.

Based on consideration of the above effects, the safety analyses performed at full power conditions are bounding for low vessel pressure conditions and even though the wide range water level instrumentation is not calibrated for low pressure conditions, this does not present a safety concern.

SIGNIFICANT HAZARD CONSIDERATION:

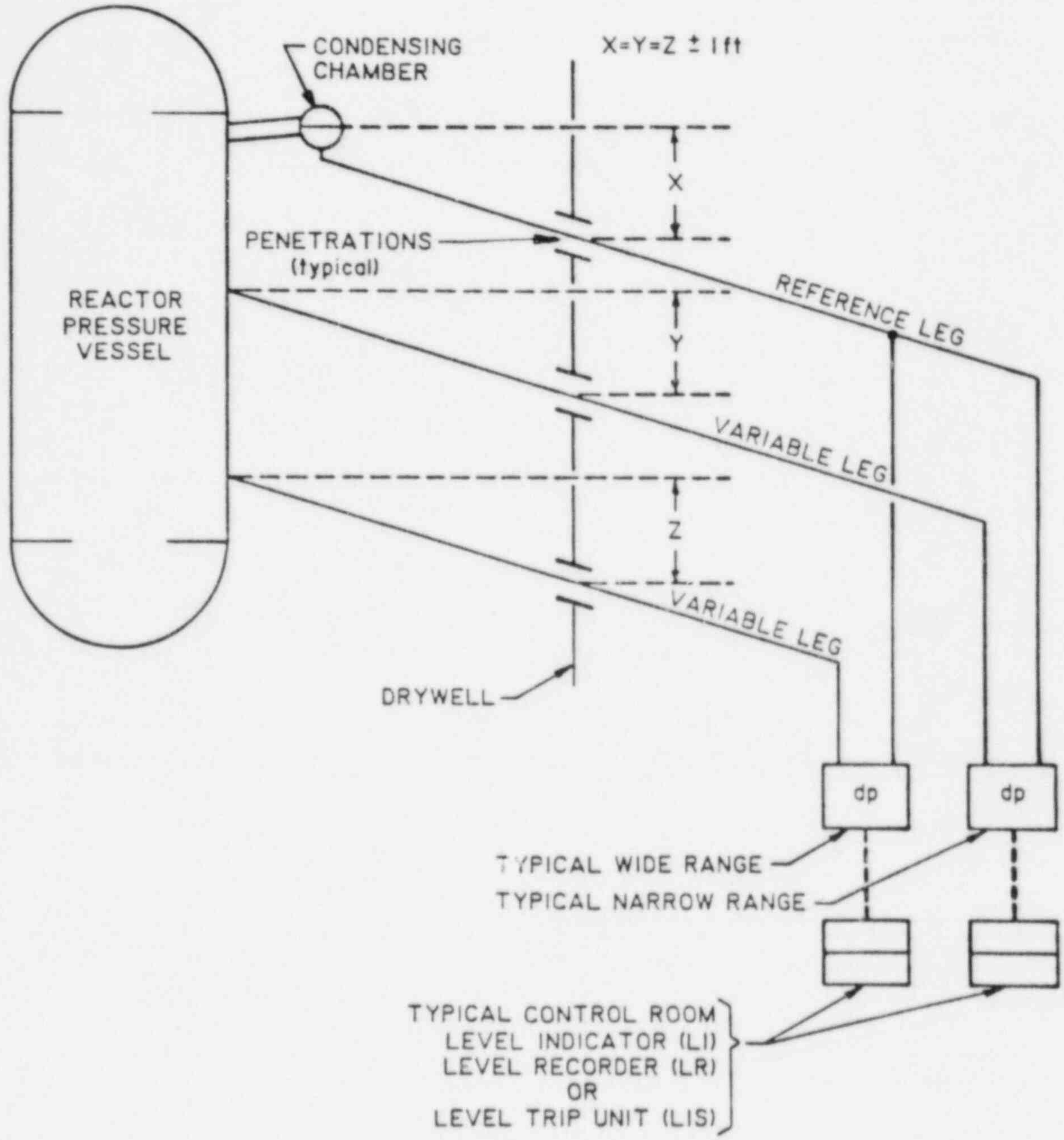
The proposed amendment does not:

1) involve a significant increase in the probability or consequences of an accident previously evaluated because the only accident analysis affected by this change is the steam line break inside the containment which was re-analyzed to show the effect of not having high drywell pressure to initiate HPCS. This event was selected because it was the only event previously in the Grand Gulf FSAR ECCS analysis which took credit for ECCS initiation on high drywell pressure. Assuming ECCS initiation on low water level and not on high drywell pressure, the LPCS D/G was found to be the most limiting failure resulting in a delay in both injection and core reflooding, and an increase of about 400°F in the calculated PCT for this event. However, since the calculated increase in the PCT is only to 1322°F, there is substantial margin to the 2200°F criteria of 10 CFR 50.46.

2) create the possibility of a new or different kind of accident from any accident previously evaluated because initiation of HPCS on low reactor water level (mass of coolant equivalent to Level 2) will occur as assumed in the present accident analysis and is not affected by this change. Since initiation of HPCS on low reactor water level is not adversely affected by this change, no new or different kind of accident from those previously analyzed are postulated to occur. The affect of not having HPCS initiate on high drywell pressure has been re-analyzed and shown to increase PCT to 1322°F which is well below the criteria of 10 CFR 50.46.

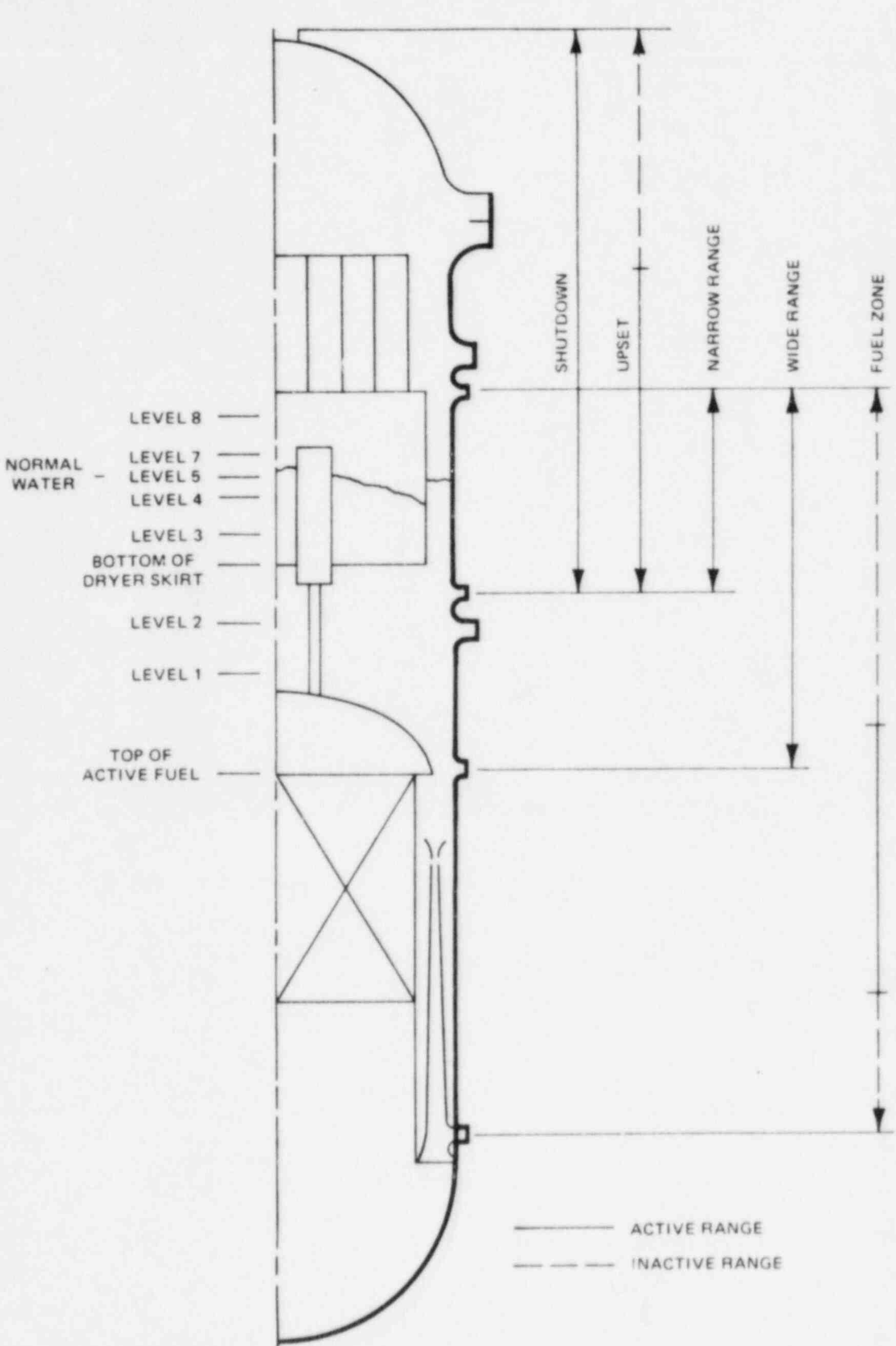
3) involve a significant reduction in a margin of safety; because, no margin of safety is affected. The only margin of safety involved is that associated with the peak cladding temperature. The criterion of 2200°F as given in 10 CFR 50.46 is met, and met with significant conservatism since the calculated increase in the PCT is only to 1322°F.

These changes make the existing requirements permanent and do not constitute a significant hazard consideration.



TYPICAL WATER LEVEL INSTRUMENTATION CONFIGURATION

Figure 1

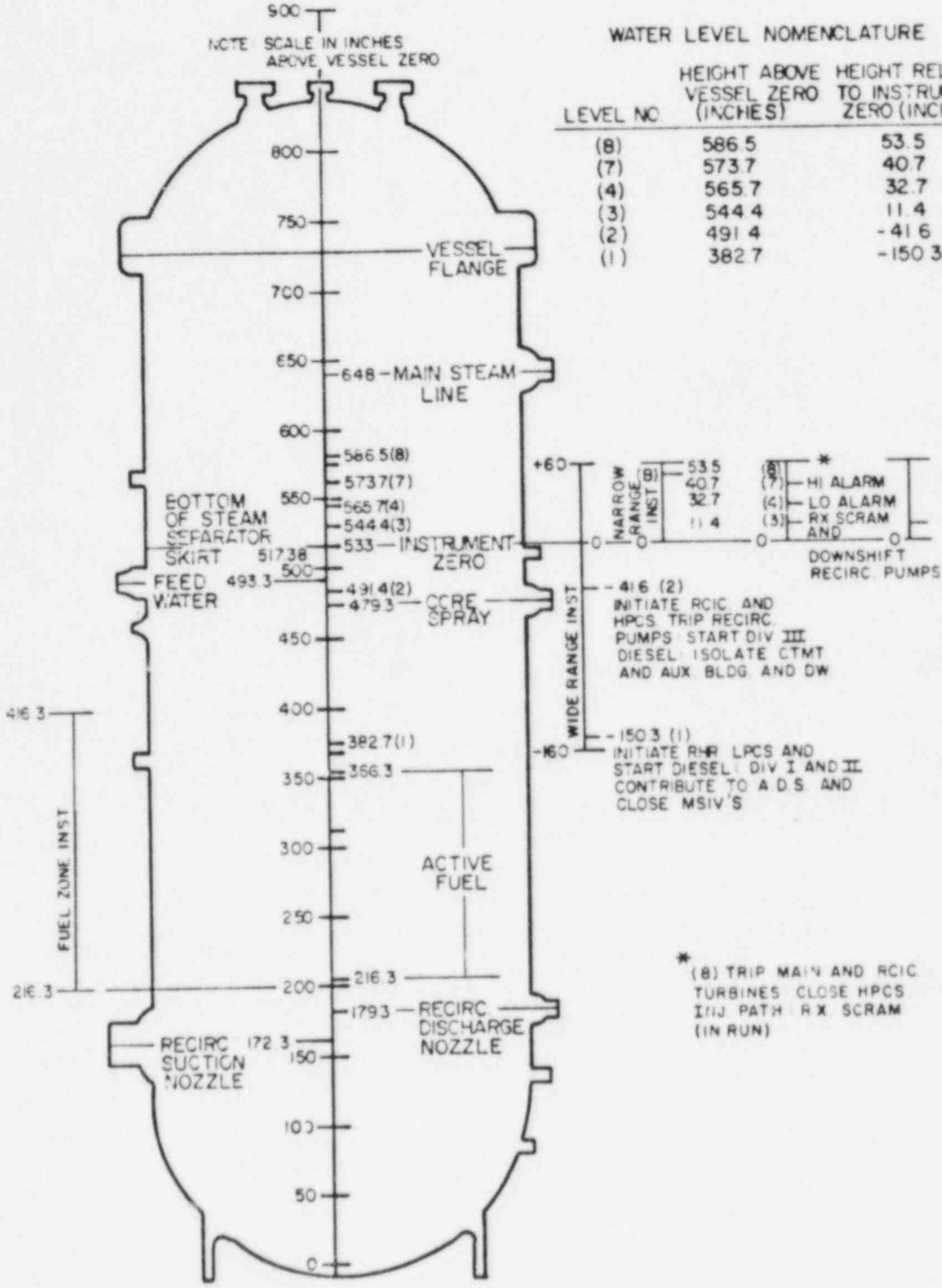


MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT	WATER LEVEL RANGE DEFINITION

Figure 2

WATER LEVEL NOMENCLATURE

LEVEL NO.	HEIGHT ABOVE VESSEL ZERO (INCHES)	HEIGHT RELATIVE TO INSTRUMENT ZERO (INCHES)
(8)	586.5	53.5
(7)	573.7	40.7
(4)	565.7	32.7
(3)	544.4	11.4
(2)	491.4	-41.6
(1)	382.7	-150.3



BASES FIGURE B 3/4 3-1 REACTOR VESSEL WATER LEVEL

Figure 3

LOCA ANALYSIS OF MAXIMUM STEAMLINE BREAK INSIDE CONTAINMENT BASED UPON LOW WATER LEVEL INITIATION

An ECCS performance evaluation was performed for Grand Gulf to determine the effect of not taking credit for ECCS initiation on high drywell pressure for a steamline break inside the containment. This assumption results in a substantial delay in the time of ECCS injections for large steamline breaks because the rapid depressurization rate initially causes a level swell. The results of this study show that the delay in ECCS injection for this event causes an increase in calculated peak cladding temperature (PCT) of about 400°F. However, this is not the limiting event in the Grand Gulf FSAR, and there is still sufficient margin to absorb this increase and remain within the 2200°F PCT limit.

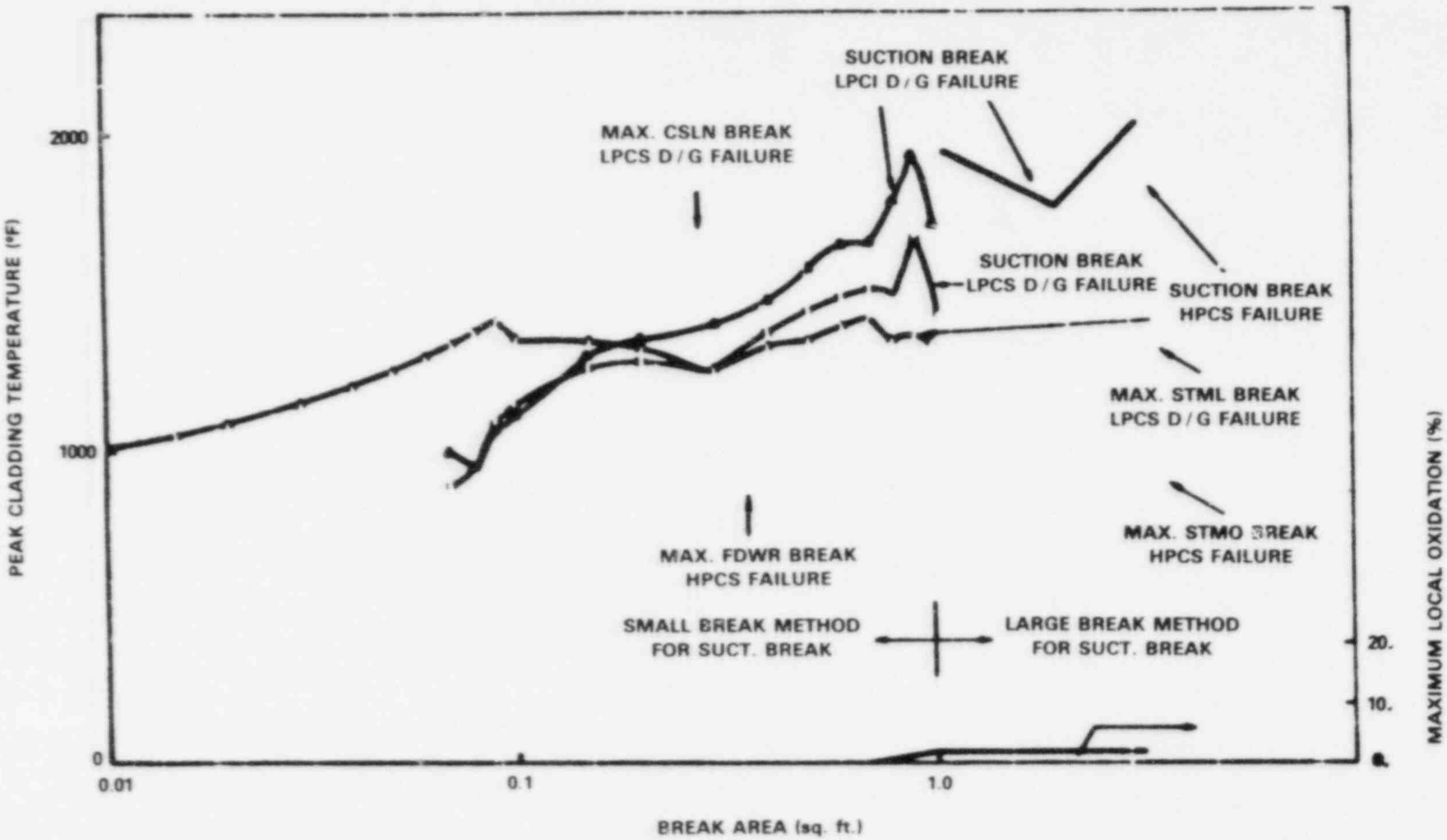
The design basis accident (DBA) steamline break inside the containment was analyzed to determine the effect of starting the ECC systems on low water level only. This event was selected because it was the only event previously in the Grand Gulf FSAR ECCS analysis which took credit for ECCS initiation on high drywell pressure.

For the previous case, the LPCI diesel generator (D/G) was the limiting failure with a calculated PCT of 934°F. The HPCS, LPCS and 1 LPCI systems were assumed to begin injecting at 27, 40 and 40 seconds respectively.

For the case with ECCS initiation on low water level, the LPCS D/G was found to be the most limiting failure with a calculated PCT of 1322°F. The HPCS and 2 LPCI systems begin injecting at 47 and 96 seconds respectively.

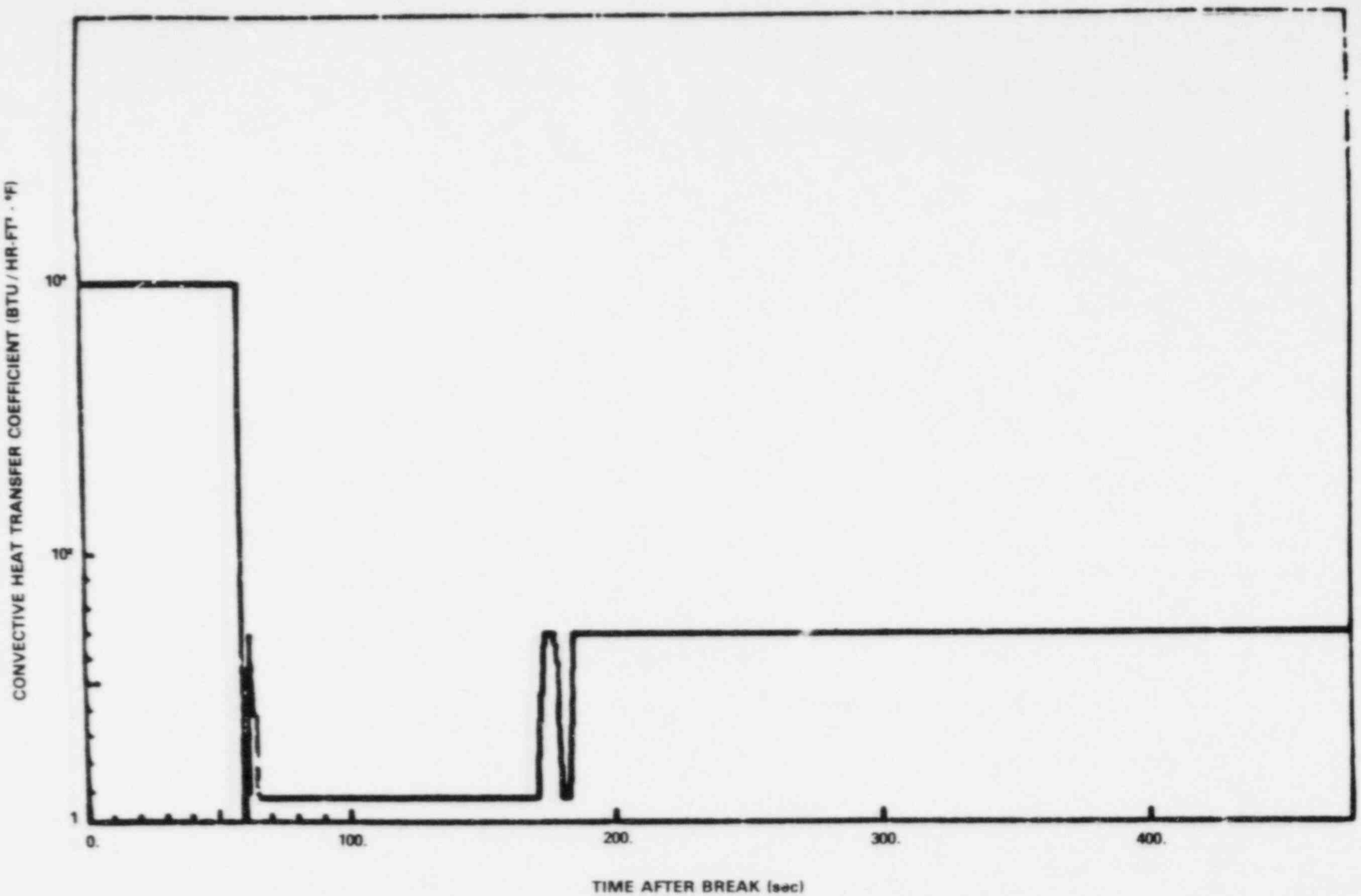
Assuming ECCS initiation on low water level results in a delay in both injection and core reflooding, and an increase of about 400°F in the calculated peak cladding temperature for this event. However, since the calculated increase in PCT is only to 1322°F, there is sufficient margin to the 2200°F limit. With these results, the Grand Gulf ECCS performance analysis conclusions are clearly not affected by the ability of the ECC systems to initiate on high drywell pressure.

To document the effect on the LOCA analysis of this change in ECCS initiation assumptions, copies of FSAR Figures 6.3-8, and 6.3-63 through 6.3-66 are attached. These figures show the PCT versus break area, and the convective heat transfer coefficient, water level, pressure and peak cladding temperature response to this event. Appropriate FSAR updates will be made to reflect the above assumption in the discussion for the main steamline break event.



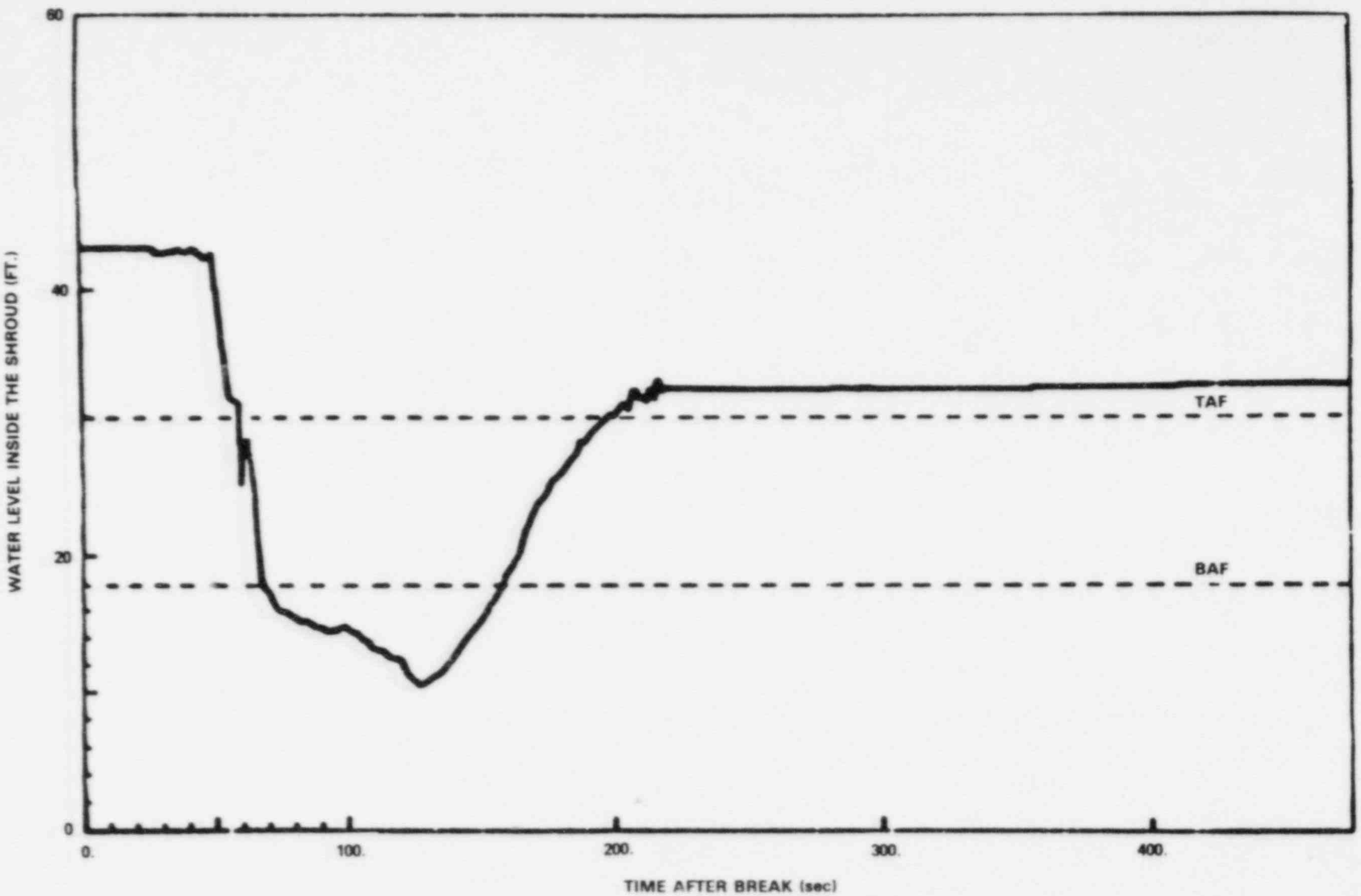
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	PEAK CLADDING TEMPERATURE AND MAXIMUM LOCAL OXIDATION VERSUS BREAK AREA FIGURE 6.3-8

Rev. 0



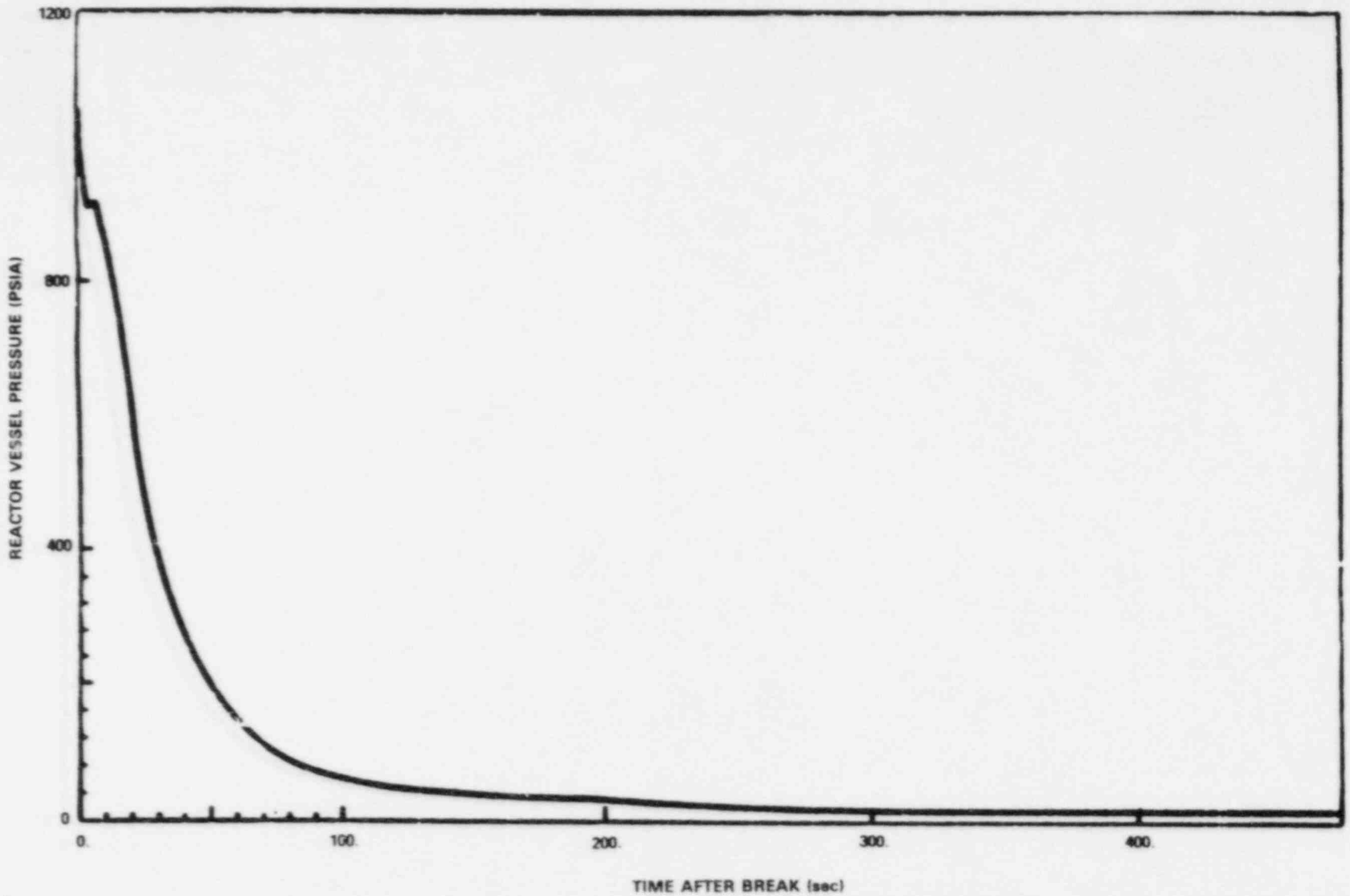
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	CONVECTIVE HEAT TRANSFER COEFFICIENT VERSUS TIME AFTER PREAK STEAMLINE BREAK (3.5 SQ. FT.), INSIDE THE CONTAINMENT, LPCS DIESEL GENERATOR FAILURE FIGURE 6.3-63

Rev. 0



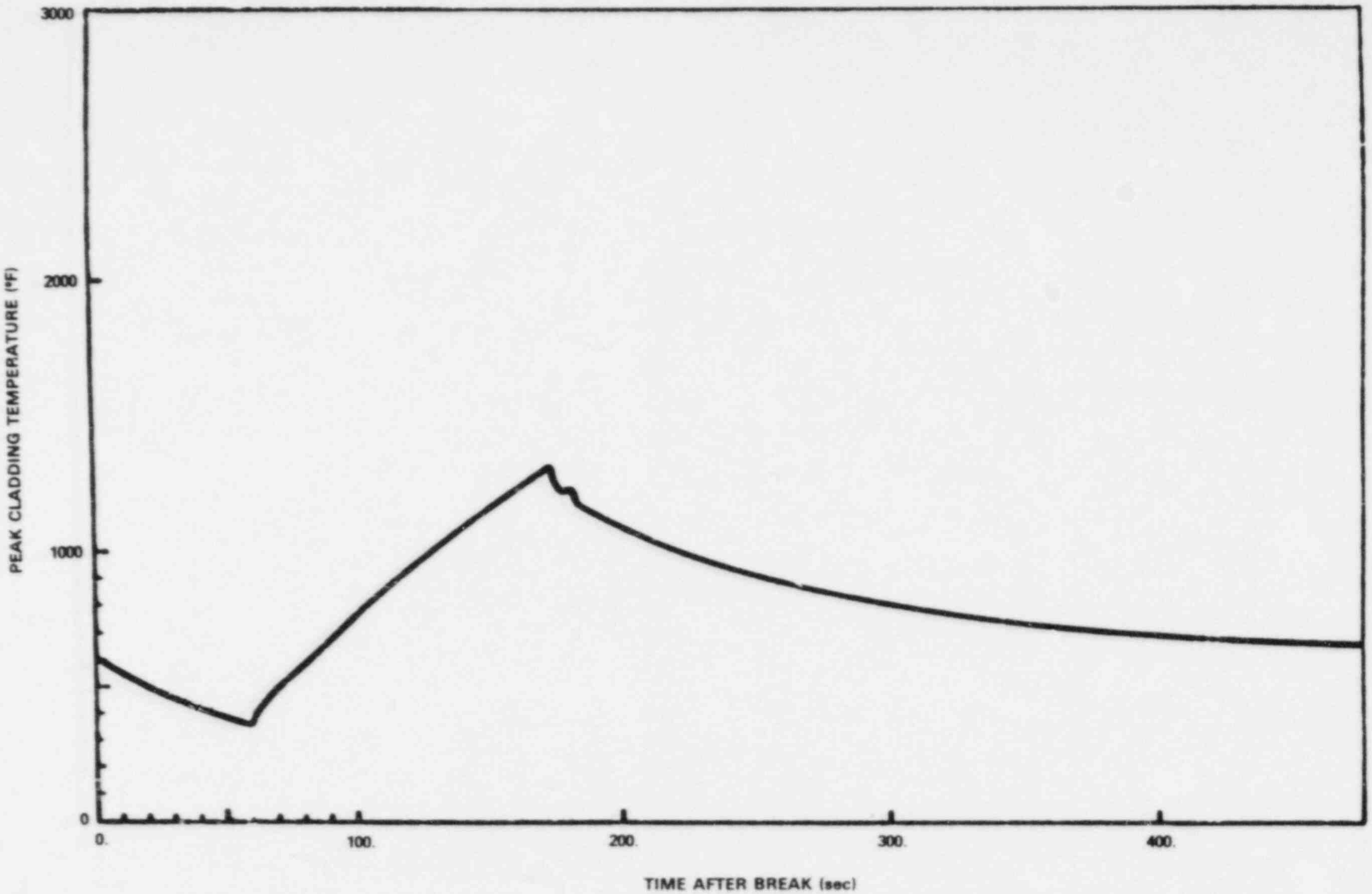
MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	WATER LEVEL INSIDE THE SHROUD VERSUS TIME AFTER BREAK STEAMLINE BREAK (3.5 SQ. FT.), INSIDE THE CONTAINMENT, LPCS DIESEL GENERATOR FAILURE FIGURE 6.3-64

Rev. 0



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	REACTOR VESSEL PRESSURE VERSUS TIME AFTER BREAK STEAMLINE BREAK (3.5 SQ. FT.), INSIDE THE CONTAINMENT, LPCS DIESEL GENERATOR FAILURE FIGURE 6.3-65

Rev. 0



MISSISSIPPI POWER & LIGHT COMPANY GRAND GULF NUCLEAR STATION UNITS 1 & 2 UPDATED FINAL SAFETY ANALYSIS REPORT	PEAK CLADDING TEMPERATURE VERSUS TIME AFTER BREAK STEAMLINE BREAK (3.5 SQ. FT.), INSIDE THE CONTAINMENT, LPCS DIESEL GENERATOR FAILURE FIGURE 6.3-66

Rev. 0

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*	33
b. Drywell Pressure - High##	4[(b)]	1, 2, 3	33
c. Reactor Vessel Water Level-High, Level 8	2[(c)]	1, 2, 3, 4*, 5*	31
d. Condensate Storage Tank Level-Low	2[(d)]	1, 2, 3, 4*, 5*	34
e. Suppression Pool Water Level-High	2[(d)]	1, 2, 3, 4*, 5*	34
f. Manual Initiation##	1	1, 2, 3, 4*, 5*	32
D. LOSS OF POWER			
1. Division 1 and 2			
a. 4.16 kV Bus Undervoltage (Loss of Voltage)	4	1, 2, 3, 4**, 5**	30
b. 4.16 kV Bus Undervoltage (BOP Load Shed)	4	1, 2, 3, 4**, 5**	30
c. 4.16 kV Bus Undervoltage (Degraded Voltage)	4	1, 2, 3, 4**, 5**	30
2. Division 3			
a. 4.16 kV Bus Undervoltage (Loss of Voltage)	4	1, 2, 3, 4**, 5**	30

(a) A channel may be placed in an inoperable status for up to 2 hours during periods of 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) Also actuates the associated division diesel generator.
(c) Provides signal to close HPCS pump discharge valve only.
(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 applicable ESF equipment is required to be OPERABLE.
\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 135 psig.
\## ~~Prior to STARTUP following the first refueling outage,~~ The injection function of Drywell Pressure - High and Manual Initiation are not required to be OPERABLE with indicated reactor vessel water level on the wide range instrument greater than Level 8 setpoint coincident with the reactor pressure less than 600 psig.

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTATION

\# Not required to be OPERABLE when reactor steam dome pressure is less than or equal to 135 psig.

\## ~~Prior to STARTUP following the first refueling outage,~~ The injection function of Drywell Pressure - High and Manual Initiation are not required to be OPERABLE with indicated reactor vessel water level on the wide range instrument greater than Level 8 setpoint coincident with the reactor pressure less than 600 psig.

* 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.

(a) Calibrate trip unit at least once per 31 days.

(b) Manual initiation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual initiation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days as a part of circuitry required to be tested for automatic system actuation.

(c) DELETED

(d) DELETED

(e) Functional Testing of Time Delay Not Required