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.

b. 4.16 kV Bus Undervoltage (Degraded Voltage)	4	1, 2, 3, 4**, 5**	30

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
D. LOSS OF POWER		
1. Division 1 and 2		
a. 4.16 kV Bus Undervoltage (Loss of Voltage)	1. 4.16 kV Basis 2912 volts	2912 +0, -291 volts
	2. 120 volt Basis 83.2 volts	83.2 +0, -8.3 volts
	3. Time Delay 0.5 seconds	0.5 +0.5, -0.1 seconds
b. 4.16 kV Bus Undervoltage (BOP Load Shed)	1. 4.16 kV Basis 3328 volts	3328 +0, -167 volts
	2. 120 volt Basis 95.1 volts	95.1 +0, -4.8 volts
	3. Time delay 0.5 seconds	0.5 +0.5, -0.1 seconds
c. 4.16 kV Bus Undervoltage (Degraded Voltage)	1. 4.16 kV Basis 3744 volts	3744 +93.6, -0 volts
	2. 120 volt Basis 107 volts	107 +2.7, -0 volts
	3. Time Delay 9.0 seconds	9.0 ± 0.5 seconds
2. Division 3		
a. 4.16 kV Bus Undervoltage (Loss of Voltage)	1. 4.16 kV Basis 3045 volts	3045 ± 61 volts
	2. 120 volt Basis 87 volts	87 ± 1.7 volts
	3. Time Delay 2.3 seconds	2.3 + 0.2, -0.3 seconds
b. 4.16 KV Bus Undervoltage (Degraded Voltage)	1. 4.16 KV Basis 3661 Volts	3661 ± 102.5 volts
	2. 120 Volt Basis 104.6 Volts	104.6 ± 2.93 volts
	3. Time Delay 5 minutes / No LOCA 4 seconds / LOCA	5 minutes ± 30 seconds (4.0 ± 0.4 seconds)

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

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
B. DIVISION 2 TRIP SYSTEM (Continued)				
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"#				
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. LPCI Pump B and C Discharge Pressure-High	S	M	R(a)	1, 2, 3
f. Manual Initiation	NA	R(b)	NA	1, 2, 3
C. DIVISION 3 TRIP SYSTEM				
1. HPCS SYSTEM				
a. Reactor Vessel Water Level - Low Low, Level 2	S	M	R(a)	1, 2, 3, 4*, 5*
b. Drywell Pressure-High##	S	M	R(a)	1, 2, 3
c. Reactor Vessel Water Level-High, Level 8	S	M	R(a)	1, 2, 3, 4*, 5*
d. Condensate Storage Tank Level - Low	S	M	R(a)	1, 2, 3, 4*, 5*
e. Suppression Pool Water Level - High	S	M	R(a)	1, 2, 3, 4*, 5*
f. Manual Initiation##	NA	R(b)	NA	1, 2, 3, 4*, 5*
D. LOSS OF POWER				
1. Division 1 and 2				
a. 4.16 kV Bus Undervoltage (Loss of Voltage)	NA	M(e)	R	1, 2, 3, 4**, 5**
b. 4.16 kV Bus Undervoltage (BOP Load Shed)	NA	M(e)	R	1, 2, 3, 4**, 5**
c. 4.16 kV Bus Undervoltage (Degraded Voltage)	NA	M(e)	R	1, 2, 3, 4**, 5**
2. Division 3				
a. 4.16 kV Bus Undervoltage (Loss of Voltage)	NA	NA	R	1, 2, 3, 4**, 5**
b. 4.16 kV Bus Undervoltage (Degraded Voltage)	NA	NA	R	1, 2, 3, 4**, 5**

9. Verifying the diesel generator operates for at least 24 hours. Diesel generators 11 and 12 shall be loaded to greater than or equal to 5450 kW but not to exceed 5740 kW for 24 hours. Diesel generator 13 shall be loaded to greater than or equal to 3630 kW for the first 2 hours of this test and to 3300 kW during the remaining 22 hours. The generator voltage and frequency shall be 4160 ± 416 volts and 60 ± 1.2 Hz within 10 seconds after the start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test. Within 5 minutes after completing this 24-hour test, perform Surveillance Requirement 4.8.1.1.2.d.7.a).2) and b).2)*.

10. Verifying that the auto-connected loads to each diesel generator do not exceed 5740 kW for diesel generators 11 and 12 and 3300 kW for diesel generator 13.

11. Verifying the diesel generator's capability to:

    a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

    b) Transfer its loads to the offsite power source, and

    c) Be restored to its standby status.

12. Verifying that with the diesel generator operating in a test mode and connected to its bus that a simulated ECCS actuation signal~~,~~

    ~~a) For Divisions 1 and 2,~~ overrides the test mode by returning the diesel generator to standby operation.

    ~~b) For Division 3, overrides the test mode by bypassing the diesel generator automatic trips per Surveillance Requirement 4.8.1.1.2.d.8.b).~~

13. [DELETED]

14. [DELETED]

15. Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval for diesel generators 11 and 12.

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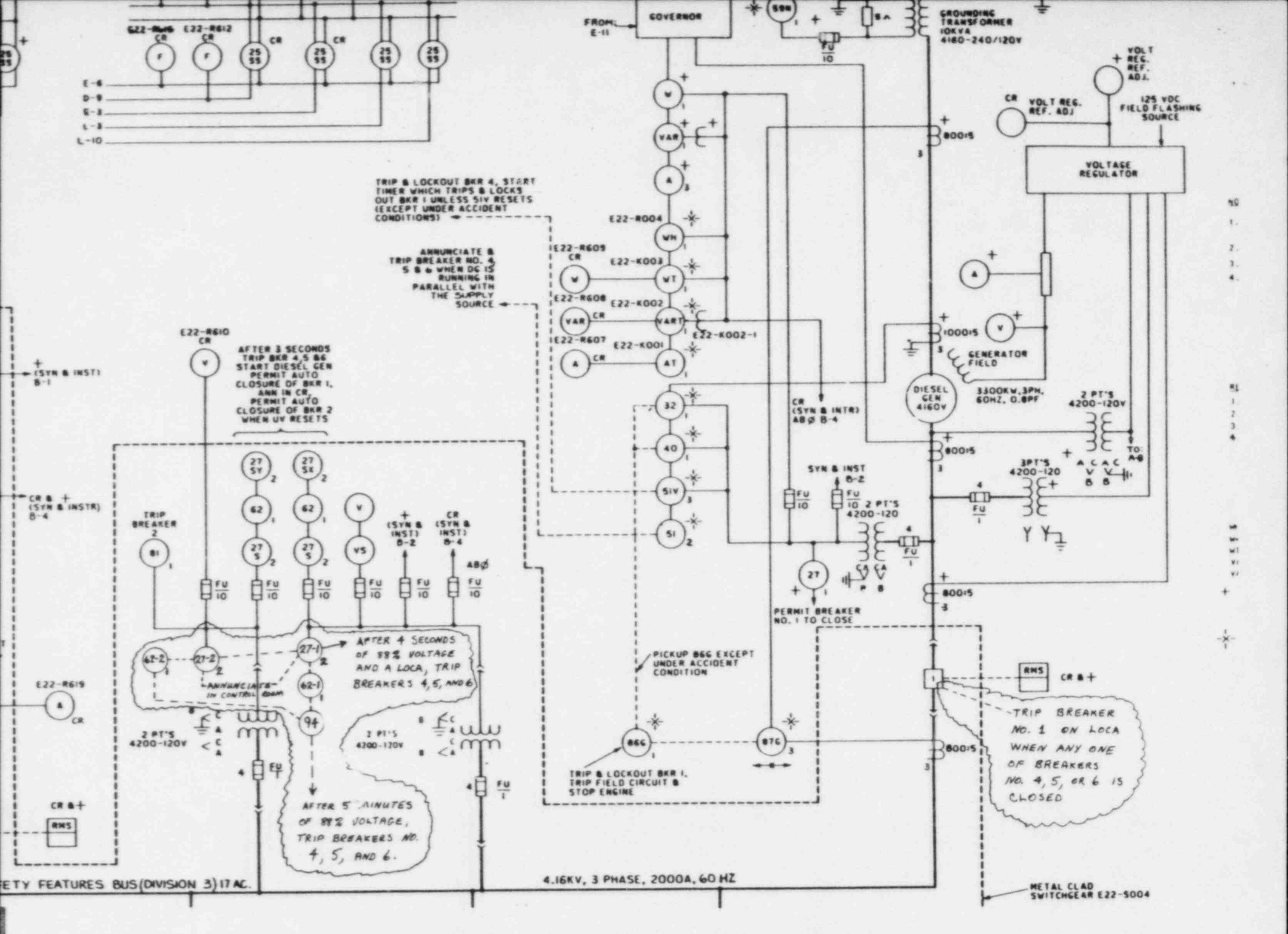
*If Surveillance Requirement 4.8.1.1.2.d.7.a)2) or b)2) are not satisfactorily completed, it is not necessary to repeat the preceding 24 hour test. Instead, the diesel generator may be operated at the load specified by Surveillance Requirement 4.8.1.1.2.a.5 for one hour or until operating temperatures have stabilized.


GOVERNOR
FROM: E-11
GROUNDING TRANSFORMER 10KVA 4160-240/120V
VOLT REG. REF. ADJ.
CR VOLT REG. REF. ADJ
125 VDC FIELD FLASHING SOURCE
VOLTAGE REGULATOR
E22-R612 CR
E-6
D-8
E-3
L-3
L-10
TRIP & LOCKOUT BKR 4, START TIMER WHICH TRIPS & LOCKS OUT BKR 1 UNLESS 51V RESETS (EXCEPT UNDER ACCIDENT CONDITIONS)
ANNUNCIATE & TRIP BREAKER NO. 4, 5 & 6 WHEN DG IS RUNNING IN PARALLEL WITH THE SUPPLY SOURCE
E22-R004
E22-R609 CR
E22-K003
E22-R608
E22-K002
E22-K002-1
E22-R607
E22-K001
800:5
1000:5
GENERATOR FIELD
DIESEL GEN 4160V
3300KW, 3PH, 60HZ, 0.8PF
2 PT'S 4200-120V
TO: A-8
3PT'S 4200-120
CR (SYN & INTR) ABØ B-4
SYN & INST B-2
2 PT'S 4200-120
(SYN & INST) B-1
CR & (SYN & INSTR) B-4
E22-R610 CR
AFTER 3 SECONDS TRIP BKR 4,5 & 6 START DIESEL GEN PERMIT AUTO CLOSURE OF BKR 1, ANN IN CR, PERMIT AUTO CLOSURE OF BKR 2 WHEN UV RESETS
TRIP BREAKER 2
(SYN & INST) B-2
CR (SYN & INST) B-4
ABØ
PERMIT BREAKER NO. 1 TO CLOSE
AFTER 4 SECONDS OF 88% VOLTAGE AND A LOCA, TRIP BREAKERS 4, 5, AND 6
ANNUNCIATE IN CONTROL ROOM
PICKUP 86G EXCEPT UNDER ACCIDENT CONDITION
E22-R619 CR
2 PT'S 4200-120V
2 PT'S 4200-120V
TRIP & LOCKOUT BKR 1, TRIP FIELD CIRCUIT & STOP ENGINE
TRIP BREAKER NO. 1 ON LOCA WHEN ANY ONE OF BREAKERS NO. 4, 5, OR 6 IS CLOSED
CR &+
RMS
AFTER 5 MINUTES OF 88% VOLTAGE, TRIP BREAKERS NO. 4, 5, AND 6.
FETY FEATURES BUS (DIVISION 3) 17 AC.
4.16KV, 3 PHASE, 2000A, 60 HZ
METAL CLAD SWITCHGEAR E22-S004

2. NPE-86/02 and OLCR-NLS-86/01

SUBJECT: Technical Specification 4.7.1.1.a.2, page 3/4 7-2
Facility Operating License No. NPF-29, pages 7 & 8

DISCUSSION: The proposed technical specification change and operating license change are the result of a design change to increase the pumping head of Standby Service Water (SSW) loop "A" which is planned for implementation during the first refueling outage now scheduled to begin in September, 1986.

It is proposed to delete the Surveillance Requirement 4.7.1.1.a.2. This surveillance requires that the SSW subsystem(s) be demonstrated operable by verifying once per 31 days that the valves isolating service to the spent fuel storage pool cooler are locked closed. Also, it is proposed to revise license condition 2.C.(20) to reflect more precise interim requirements.

As currently written, the license condition could be interpreted to prohibit placing irradiated fuel in the spent fuel pool until both loop A and loop B of the SSW system are modified. The proposed revision to the license condition will allow irradiated fuel to be placed in the spent fuel pool with one loop of the SSW system operable; i.e., prior to loop A modification. This is consistent with the approved GGNS Technical Specification 3.7.1.3. SSW Loop B has been modified and tested to assure that it is capable of providing required coolant flow to the essential SSW system components. Therefore, during SSW loop A modifications, loop B will provide the required coolant flow. The RHR system is available as a backup to the Fuel Pool Cooling and Cleanup system if required.

It is proposed that the revised license condition read as follows:

(20) Standby Service Water System (Section 9.2.1 SER, SSER #2)

No irradiated fuel may be stored in the Unit 1 spent fuel storage pool prior to completion of modifications to either loop A or loop B of the standby service water (SSW) system and verification that the design flow can be achieved to all essential SSW system components in the modified loop. However, should a core offloading be necessary prior to completion of these modifications (scheduled for the first refueling outage), irradiated fuel may be placed in the spent fuel pool when the RHR system operating in the spent fuel pool cooling mode is available. Until the SSW loops are modified, the spent fuel pool cooler in an unmodified loop shall be isolated from the loop by locked closed valves or the loop shall be declared inoperable. The position of these valves shall be verified every 31 days until the design flowrate for the SSW loop is demonstrated. The surveillance to be performed is to verify that any unmodified SSW loop with valves which are not locked closed is declared inoperable.

JUSTIFICATION: The SSW system consists of three independent piping loops labeled "A", "B", and "C". The "A" and "B" loops serve redundant plant equipment. The "C" loop serves only the HPCS pump and its associated diesel generator and support systems. The redundant plant equipment served by loops "A" and "B" includes spent fuel pool coolers "A" and "B" respectively. During startup testing on the SSW system at Grand Gulf, flow tests indicated that certain essential equipment would not receive full design flow. Consequently, a corrective action taken by MP&L on an interim basis was to isolate the spent fuel pool coolers from each of the SSW system loops "A" and "B" in order to provide increased flow to the other components in each of the loops. As a permanent solution to the low flow conditions, MP&L committed to provide larger SSW pump motors and impellers prior to or during the first refueling outage.

As a result, the operating license was changed to incorporate a condition restricting the storage of fuel in the spent fuel pool until completion of the modifications to the SSW system. The license condition and Surveillance Requirement 4.7.1.1.a.2 specified that the fuel pool cooler shall be isolated from the SSW system by locked closed valves and the position of the valves are to be verified every 31 days until the design flowrate for the SSW system is demonstrated.

License condition 2.C.(20), as written, requires modification of both "A" and "B" loops of the SSW system in order to obtain compliance. MP&L completed the required modifications to the "B" loop of the SSW system during an outage in the fall of 1985. Technical Specification changes associated with this modification were submitted to the Nuclear Regulatory Commission (NRC) and were approved by the NRC in a letter to MP&L dated October 12, 1985.

The modified Loop "B" SSW pump has been tested and demonstrated to be capable of providing the required flow to the fuel pool cooler (Fuel Pool Heat Exchanger Q1G41B001B) as well as the remaining essential components served by loop "B" of the SSW system. Therefore, isolation of the SSW loop "B" fuel pool cooler is no longer necessary. Also, the attached SSW RF01 analysis summary shows that for the first refueling outage, SSW basin B meets the design requirement of containing a 30 day supply of water without makeup. This requirement is met for the first refueling outage without the necessity of an operable siphon between the basins.

Following the modification to the "A" loop of the SSW system, the fuel pool cooler (Fuel Pool Heat Exchanger Q1G41B001A) will no longer need to be isolated. Both modified loops will be capable of providing the required flow to the fuel pool coolers along with the other essential components served by each of the loops. Therefore, it is proposed that license condition 2.C.(20) be amended to reflect the more precise requirements and to become effective immediately. It is also proposed that the deletion of the Surveillance Requirement 4.7.1.1.a.2 of the Technical Specification be effective upon shutdown of the reactor for the first refueling outage at Grand Gulf 1. Deletion of the surveillance requirement is justified because the proposed license condition includes an identical operability limitation for an unmodified loop. The license condition is adequate to control operability without its corresponding technical specification while loop A is being modified during RF01. Further, during the modification of loop A, loop B will provide the required cooling capacity, and the RHR system will be available as a backup to the Fuel Pool Cooling and Cleanup system if needed.

The license amendment and deletion of the surveillance requirement do not involve any significant hazards considerations as discussed below; however, the changes are significant in that they are necessary to permit fuel movement during the early stage of the refueling outage. Early fuel movement is necessary in order to complete an extensive inservice inspection of the reactor pressure vessel in a timely manner. If fuel movement is not accomplished early into the refueling outage, the critical path of the outage will be extended by as much as 7 days.

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 proposed change to the Technical Specifications deletes Surveillance Requirement 4.7.1.1.a.2 since Fuel Pool Heat Exchanger Q1G41B001A will no longer need to be locked out after the modifications to Loop "A" of the SSW system. After these modifications are completed, both of the SSW loops will be capable of providing the required flow to the respective fuel pool coolers and other essential components. Also, with an operable siphon, either basin will be capable of providing a 30 day water supply without makeup to the SSW system. The proposed change to the operating license condition 2.C.(20) simply provides clarification by more precisely reflecting the requirements. Specifically, with one loop of the SSW system operable, irradiated fuel may be placed in the spent fuel pool in accordance with the governing Technical Specifications.

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because the modifications to Loop "A" along with the already completed modifications to Loop "B" ensures that both loops of the SSW system will perform as designed for all previously evaluated accidents. The design change will ensure that the required coolant flowrate is provided to the essential components of each of the SSW loops; and, with an operable siphon, either basin will meet the design requirement of having a 30 day self-contained supply of water without makeup. Therefore, this change cannot increase the probability or consequences of an accident.

The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because during the modifications to Loop "A", SSW Loop "B" will be operable in accordance with governing Technical Specifications. The modifications completed during the Fall 1985 Outage for Loop "B" ensure that this loop of the SSW system will provide all essential system components with the required flowrate. Further, the RHR system is available as a backup to the Fuel Pool Cooling and Cleanup system if required. Upon completion of the modifications to Loop "A", both SSW loops will be capable of providing the required coolant flow rate and be capable of utilizing the required heat removal capacity of the basins. The improved capabilities of both SSW loops does not create the possibility of a new or different kind of accident.

The proposed changes do not involve a significant reduction in a margin of safety because the design modifications are for the purpose of ensuring that both loops of the SSW system will perform as designed. The increased head of the new SSW pumps will ensure that both loops of the SSW system are capable of providing design flowrates to all essential system components. While modifications are performed on loop A of the SSW system, the plant will be in a shutdown condition with loop B of the SSW providing the required cooling for the spent fuel pool and with the RHR system available as a backup to the Fuel Pool Cooling and Cleanup system if required.

Therefore, the proposed changes involve no significant hazards considerations.

SSW RF01 ANALYSIS SUMMARY

Regulatory Guide 1.27 requires that the Ultimate Heat Sink (UHS) be capable of providing sufficient cooling water to dissipate residual heat after a design basis accident (DBA) for a period of 30 days without inventory replenishment. To assure this requirement is met, a Standby Service Water (SSW) system analysis was performed to verify the capability of the SSW UHS to dissipate the residual heat associated with a DBA in Unit 1 with Unit 2 not in operation. Analysis results indicated that the total evaporative and drift losses from the UHS cooling tower for the 30 day period following the DBA in Unit 1 exceed the total usable volume of a single basin, necessitating the use of basin transfer capabilities (basin siphon).

During the First Refueling Outage (RF01), however, SSW Basin A will be taken out of service and drained to allow modification of the loop A SSW pump and piping. Therefore to satisfy the requirements of Regulatory Guide 1.27 without basin transfer capabilities, an RF01 specific SSW system analysis has been performed to verify the capability of the SSW UHS to dissipate the residual heat associated with a DBA in Unit 1 with Unit 2 not in operation and with SSW Basin A drained.

The system analysis is based on the following assumptions:

1) Unit 1 is experiencing a loss of coolant accident (LOCA) coincident with a loss of offsite power (LOP),

2) Unit 2 is not operational,

3) SSW loop A is removed from service and SSW Basin A is drained (Inventory transfer between Basin A and Basin B is not possible),

4) Makeup to the SSW system is not available from normal means of supply, and

5) Worst 30 day meteorology per Regulatory Guide 1.27.

6) The spent fuel pool contains no fuel from previous discharges.

These assumptions will result in the greatest heat rejection rate for the UHS during the most severe meteorology following the DBA for the RF01 scenario. These assumptions are conservative since fuel movement to the spent fuel pool is not presently scheduled to begin until approximately 5 days after reactor shutdown.

ANALYSIS RESULTS - Analysis results indicate the total water losses from the SSW UHS cooling tower will be 6,209,798 gallons for the 30 day post LOCA period. The total usable Basin B volume from elevation 84'-6" to elevation 130'-3" is 6,638,508 gallons. With the total water loss, evaporative and drift, calculated to be 6,209,798 gallons, there will be 428,710 gallons remaining at the end of the 30 day period. Therefore, the RF01 specific performance analysis verifies the 30 day post LOCA inventory requirement. The analysis also verifies the capability of the UHS to dissipate the residual heat since that even with a gradual depleting basin water inventory due to evaporation and drift, the maximum cold water temperature of 86.1^{o}F will still not exceed the design temperature of 90^{o}F.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

d. In OPERATIONAL CONDITION 5 with the SSW subsystem, which is associated with an RHR system required OPERABLE by Specification 3.9.11.1 or 3.9.11.2, inoperable, declare the associated RHR system inoperable and take the ACTION required by Specification 3.9.11.1 or 3.9.11.2, as applicable.

e. In OPERATIONAL CONDITION *, with the SSW subsystem, which is associated with a diesel generator required OPERABLE by Specification 3.8.1.2, inoperable, declare the associated diesel generator inoperable and take the ACTION required by Specification 3.8.1.2. The provisions of Specification 3.0.3 are not applicable.

f. In OPERATIONAL CONDITIONS 1, 2, 3, 4, or 5 with the SSW subsystem, which is associated with a diesel generator required OPERABLE by Specification 3.8.1.1 or 3.8.1.2, inoperable, declare the associated diesel generator inoperable and take the ACTION required by Specification 3.8.1.1 or 3.8.1.2 as applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 At least the above required standby service water system subsystem(s) shall be demonstrated OPERABLE:

a. At least once per 31 days by:

   1. Verifying that each valve in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.
   ~~2. Verifying that the valves isolating service to the spent fuel storage pool cooler are locked closed.~~

b. At least once per 18 months during shutdown by verifying that each automatic valve servicing safety related equipment actuates to its correct position on an actuation test signal.

(15) Scram Discharge Volume (Sections 4.6, SER)

Prior to startup following the first refueling outage, MP&L shall incorporate the following additional modifications into the scram discharge volume system:

(i) Redundant vent and drain valves, and

(ii) Diverse and redundant scram instrumentation for each instrumented volume, including both delta pressure sensors and float sensors.

(16) Containment Purge (Section 6.2.4, SSER #5)

Prior to startup following the first refueling outage, MP&L shall provide for NRC review a reevaluation of the need to use the containment purge mode of the containment cooling system. This study should include, but is not limited to, data gathered during the first fuel cycle related to airborne activity level (ALARA), overall containment air quality and personnel access to containment. Based on the above cited study, MP&L shall propose the purge criteria to be used for the remainder of the plant life.

(17) Containment Pressure Boundary (Section 6.2.8, SER)

Prior to startup following the first refueling outage, MP&L shall replace the feedwater check valve disc with a disc made from a suitable material.

(18) Pressure Interlocks on Valves Interfacing at Low and High Pressure (Section 6.3.4, SSER #2)

Prior to startup following the first refueling outage, the licensee shall implement isolation protection against overpressurization of the low pressure emergency core cooling systems (RHR/LPCI and LPCS) at the high and low pressure interface containing a check valve and a closed motor-operated valve.

(19) IE Information Notice 79-22, Qualification of Control System (Section 7.8.C, SER, SSER #2)

Prior to startup following the first refueling outage, MP&L shall complete any design changes found necessary as a result of this review.

(20) Standby Service Water System (Section 9.2.1 SER, SSER #2)

~~No irradiated fuel may be stored in the Unit 1 spent fuel pool prior to completion of modifications to the standby service water~~

INSERT →

~~(SSW) system and verification that the design flow can be achieved to all SSW system components. However, should a core offloading be necessary prior to completion of these modifications (scheduled for the first scheduled refueling outage), irradiated fuel may be placed in the spent fuel pool when the RHR system operating in the spent fuel pool cooling mode is available. Until the SSW system is modified, the spent fuel pool cooler shall be isolated from the SSW system by locked closed valves or the associated SSW subsystem shall be declared inoperable. The position of these valves shall be verified every 31 days until the design flowrate for SSW system is demonstrated. The surveillance to be performed is to verify that any SSW loop with valves which are not locked closed is declared inoperable.~~

AMEND. No. 5

(21) Spent Fuel Pool Ventilation System (Section 9.4.2, SER, SSER #2)

If spent irradiated fuel is placed in the spent fuel pool prior to installation and operability of the safety related backup fuel pool cooling pump room coolers, the plant shall be placed in shutdown condition and remain shutdown with the RHR system dedicated to the fuel pool cooling mode.

(22) Remote Shutdown Panel (Section 9.5.4.1, SER, SSER #2)

Prior to startup following the first refueling outage, MP&L shall install electrical isolation switches between the control room and the Division 1 remote shutdown panel.

(23) Fire Protection Program (Section 9.5.9, SER)

MP&L shall maintain in effect and fully implement all provisions of the approved Fire Protection Plan. In addition, MP&L shall maintain the fire protection program to meet the intent of Appendix R to 10 CFR Part 50, except that an oil collection system for the reactor coolant pump is not required.

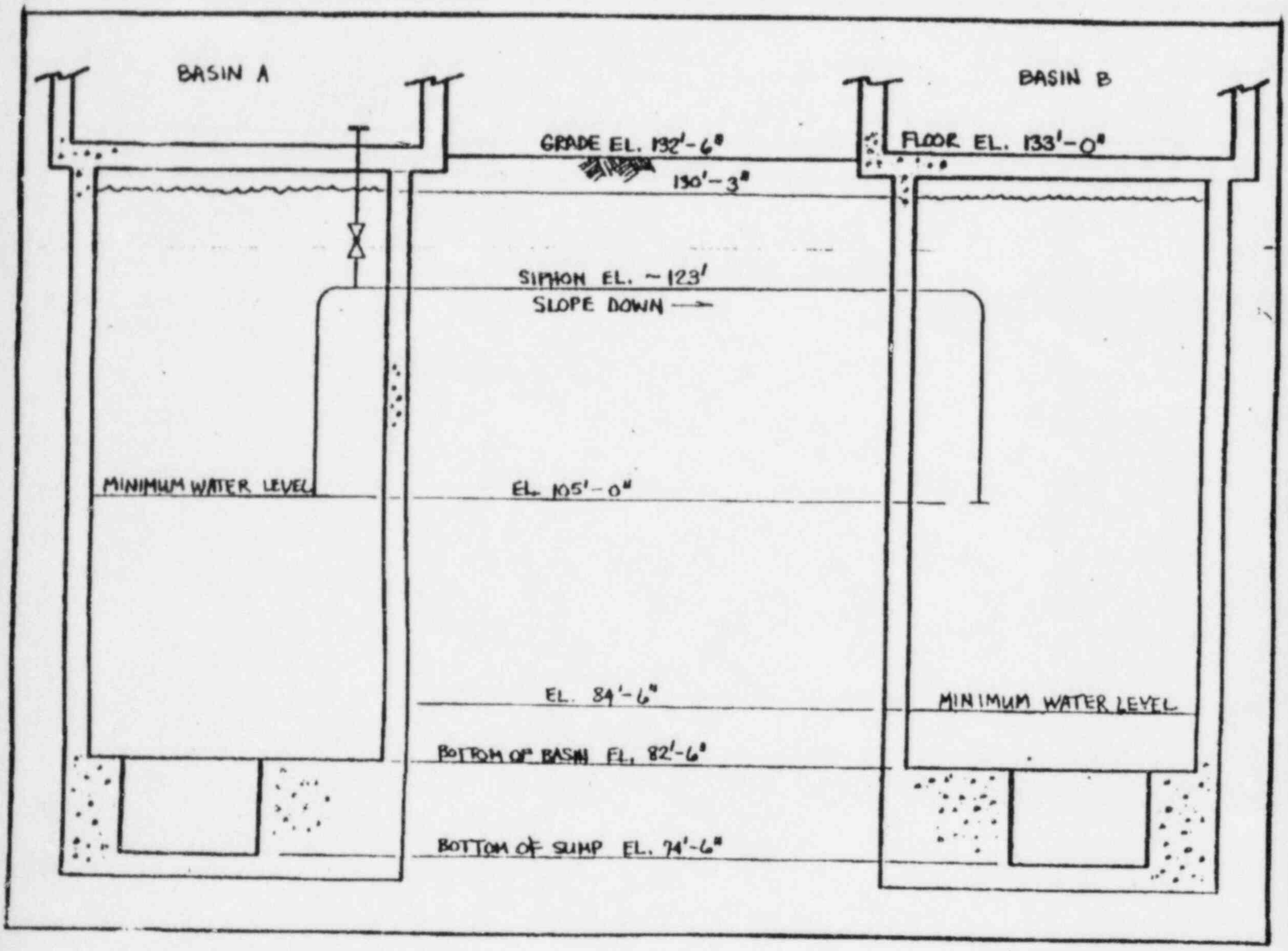
(24) Interplant Communication Systems (Section 9.6.1.2, SER, SSER #2, SSER #4, SSER #5)

Tests of the communication systems used to mitigate the consequences of an event and attain a safe plant shutdown shall be completed during preoperational and startup tests. An evaluation of the test results shall be provided for NRC review within 90 days after test completion. Any system modifications found necessary as a result of NRC review shall be completed prior to startup following the first refueling outage.

INSERT TO LICENSE CONDITION 2.C.(20).

(20) Standby Service Water System (Section 9.2.1 SER, SSER #2)

No irradiated fuel may be stored in the Unit 1 spent fuel storage pool prior to completion of modifications to either loop A or loop B of the standby service water (SSW) system and verification that the design flow can be achieved to all essential SSW system components in the modified loop. However, should a core offloading be necessary prior to completion of these modifications (scheduled for the first refueling outage), irradiated fuel may be placed in the spent fuel pool when the RHR system operating in the spent fuel pool cooling mode is available. Until the SSW loops are modified, the spent fuel pool cooler in an unmodified loop shall be isolated from the loop by locked closed valves or the loop shall be declared inoperable. The position of these valves shall be verified every 31 days until the design flowrate for the SSW loop is demonstrated. The surveillance to be performed is to verify that any unmodified SSW loop with valves which are not locked closed is declared inoperable.


BASIN A
BASIN B
GRADE EL. 132'-6"
FLOOR EL. 133'-0"
130'-3"
SIPHON EL. ~123'
SLOPE DOWN
MINIMUM WATER LEVEL
EL. 105'-0"
EL. 84'-6"
MINIMUM WATER LEVEL
BOTTOM OF BASIN EL. 82'-6"
BOTTOM OF SUMP EL. 74'-6"

3. PS-86/04

SUBJECT: Technical Specification 4.3.4.2.3, page 3/4 3-42; Bases For Specification 3/4.3.4, page B 3/4 3-3; and Definition 1.13, page 1-3.

DISCUSSION: The proposed changes provide clarification of the terminology and intent of Surveillance Requirement 4.3.4.2.3 for the End-of-Cycle Recirculation Pump Trip (EOC-RPT) system response time. As currently written, the specification requires verification of an Arc Suppression time of 50 msec while the manufacturer had determined by test that the actual Arc Suppression time for the Grand Gulf breakers is 12 msec.

The intent of Technical Specification 4.3.4.2.3 is clarified in a footnote by defining "Breaker Arc Suppression" as a part of "Breaker Interruption" time. Arc Suppression is not physically measured; instead, the manufacturer's design value is validated by performance of contact gap measurements and high potential tests on the breaker vacuum interrupters. This change also requires the manufacturer's maximum design Arc Suppression time of 12 msec to be added to the tested Breaker Response time to determine the Breaker Interruption time. These changes have no effect on the overall response time requirements (190 msec) of Technical Specification 4.3.4.2.3 and Table 3.3.4.2-3.

JUSTIFICATION: This change is required to clarify the terminology used in the GGNS Technical Specifications. The existing terminology "Breaker Arc Suppression Time" was intended to be "Breaker Interruption Time" which, as defined by General Electric Design Specifications for the Reactor Recirculation System, is the time interval from application of voltage to the trip coil to the complete suppression of the electrical arc across the open breaker contacts. The Interruption time consists of "Breaker Response Time" (the time from application of voltage to the trip coil until main contacts separate) and "Arc Suppression Time" (the time from main contact separation until the complete suppression of the electrical arc across the open contacts).

The existing 60 month test requirement to determine breaker arc suppression time is impractical to perform due to the nature of the test. The test would require two high speed trips for each Reactor Recirculation Pump at pump speeds above 40%, thus subjecting the plant to additional undesired transients. Alternate methods using dummy loads to test the breakers would require extensive plant modifications. To obtain a truly representative arc, a load of about 650 amps at 6900 volts would be required to realistically represent the electrical conditions, so that the arc suppression time could be determined.

Based on General Electric design and production testing performed on this type of breaker (GE model VB-7.2-500-12A "Power Vac") if a high potential test of the vacuum interrupter is satisfactory, the design value for breaker arc suppression time will be valid. Electrical surveillance procedure 06-EL-1B33-0-0001 provides for a high potential test on the vacuum interrupter, it also provides for the measurement of contact gap, and it provides for other recommended maintenance and testing described in GE vendor manual GEK-396711-D. This maintenance and testing is performed at least once per 60 months as required by Technical Specification 4.8.4.1.b.

A timing test for arc suppression is neither described nor recommended by GEK-39671-D. Since vacuum interrupter integrity and contact gap measurements are the primary factors for establishing the design arc suppression time of 12 msec, a timing test for arc suppression is not necessary.

The manufacturer, General Electric, supports the technical specification change and has stated that assurance of arc suppression time as herein outlined when coupled with measurement of the other response time components will provide an overall response time for the EOC-RPT function that is fully compatible with the plant safety analysis.

The proposed change to Technical Specification 4.3.4.2.3 would require the response time of the breaker, approximately 35 to 45 msec, to be tested during the normal 5 year maintenance surveillance with the plant in a cold shutdown condition. Changes to the Bases reflect the changes made to Technical Specification 4.3.4.2.3 and further clarify the bases.

SIGNIFICANT HAZARD CONSIDERATION:

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because there are no changes to the existing plant design and no new failure modes are created. Also, the ability of the EOC-RPT system to respond within 190 msec is not affected. Existing surveillance data indicates a worst case overall response time of 141 msec; thus, more than adequate tolerance to the limit exists. A "high potential" test of the Recirculation Pump breaker vacuum interrupters adequately ensures proper breaker function including arc suppression. Existing surveillance data also shows ample margin to the overall 190 msec value, such that use of the manufacturer's maximum design value of 12 msec for arc suppression does not represent any degradation to the validity of the determination of the overall EOC-RPT system response time. A 60 month interval on measurement of "Breaker Response time" adequately ensures proper breaker operation.

The proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated because there are no changes to the existing plant design and the required total EOC-RPT system response time of equal to or less than 190 msec remains unchanged. No new measurement techniques are introduced. The proposed change simply provides clarification of surveillance requirements and makes permanent the use of a constant value for breaker arc suppression as currently required by the Technical Specifications. The changes make it clear that the controlling factor for assuring safety is total system response time and that the actual arc suppression time is validated by measurement of the controlling parameters. The changes have no impact on the total system response time which remains unchanged at 190 msec.

The proposed changes do not involve a significant reduction in a margin of safety. No margins of safety are impacted. The proposed changes simply clarify the surveillance requirements which will provide for verification of the functional response time of the EOC-RPT system. Existing surveillance data indicates a worst case total response time of 141 msec and this is based on using the maximum design value for arc suppression. Thus, more than adequate tolerance to the limit exists for the total system response time which remains unchanged at 190 msec.

Ē-AVERAGE DISINTEGRATION ENERGY

1.11 Ē shall be the average, weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling, of the sum of the average beta and gamma energies per disintegration, in MeV, for isotopes, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME

1.12 The EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS actuation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function, i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc. Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME

1.13 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME shall be that time interval to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker from initial movement of the associated:

a. Turbine stop valves, and
b. Turbine control valves.

The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured,

FRACTION OF LIMITING POWER DENSITY

1.14 The FRACTION OF LIMITING POWER DENSITY (FLPD) shall be the LHGR existing at a given location divided by the limiting LHGR for that bundle type.

FRACTION OF RATED THERMAL POWER

1.15 The FRACTION OF RATED THERMAL POWER (FRTP) shall be the measured THERMAL POWER divided by the RATED THERMAL POWER.

FREQUENCY NOTATION

1.16 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GASEOUS RADWASTE TREATMENT (OFFGAS) SYSTEM

1.17 The GASEOUS RADWASTE TREATMENT (OFFGAS) SYSTEM is the system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

except for the breaker arc suppression time which is not measured but is validated by surveillance tests to conform to the manufacturer's design value.

SURVEILLANCE REQUIREMENTS

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

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

4.3.4.2.3 The END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME of each trip function shown in Table 3.3.4.2-3 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include two turbine control valve channels from one trip system and two turbine stop valve channels from the other trip system such that all channels are tested at least once per 36 months. ~~The time allotted for breaker arc suppression, 50 ms, shall be verified at least once per 60 months.*~~

The time required for Breaker Interruption* shall be verified at least once per 60 months and added to the 18 month trip system times to verify that the overall END-OF-CYCLE RECIRCULATION PUMP TRIP RESPONSE TIME is within its limit.

~~*Prior to STARTUP after the first refueling outage, the breaker arc suppression time of 12 ms, as determined by the manufacturer, shall apply.~~

* Breaker Interruption time is defined as: Breaker Response time plus Arc Suppression time. Breaker Response is the time from application of voltage to the trip coil until the main contacts separate. Arc Suppression is the time from main contact separation until the complete suppression of the electrical arc across the open contacts. Breaker Response shall be verified by testing and added to the manufacturer's design Arc Suppression time of 12 ms to determine Breaker Interruption time. The breaker arc suppression time shall be validated by the performance of periodic contact gap measurements and high potential tests on the breaker vacuum interrupters in accordance with plant procedures.

Included in this time are: the response time of the sensor, the response time of the system logic and the breaker interruption time. Breaker interruption time includes both breaker response time and the manufacturer's design arc suppression time of 12 ms.

RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION (Continued)

feature will function are closure of the turbine stop valves and fast closure of the turbine control valves.

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

Each EOC-RPT system may be manually bypassed by use of a keyswitch which is administratively controlled. The manual bypasses and the automatic Operating Bypass at less than 40% of RATED THERMAL POWER are annunciated in the control room.

The EOC-RPT system response time is the time assumed in the analysis between initiation of valve motion and complete suppression of the electric arc; i.e., 190 ms, ~~less the time allotted from start of motion of the stop valve or turbine control valve until the sensor relay contact supplying the input to the reactor protection system opens, i.e., 70 ms, and less the time allotted for breaker arc suppression determined by test, as correlated to manufacturer's test results, i.e., 50 ms, and plant pre-operational test results.~~

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or greater than the drift allowance assumed for each trip in the safety analyses.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

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

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or greater than the drift allowance assumed for each trip in the safety analyses.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

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

The OPERABILITY of the control rod block instrumentation in OPERATIONAL CONDITION 5 is to provide diversity of rod block protection to the one-rod-out interlock.

4. NPE-86/11

SUBJECT: Technical Specification Surveillance Requirement 4.8.2.1.d.2; page 3/4 8-12

DISCUSSION: The change to the subject specification is proposed to reflect new loads to be added to Division 1 and Division 2 batteries at the first refueling outage and to reflect the maximum load rating of the uninterruptable power supplies (UPS).

The design changes that will increase the loading on the Division 1 and Division 2 batteries are scheduled for implementation not later than startup from 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: Present Technical Specification Surveillance Requirement 4.8.2.1.d.2 specifies an 18 month testing requirement to help ensure that battery capacity is greater than the actual emergency load while maintaining the battery terminal voltage greater than or equal to 105 volts. The present 125 volt DC Division 1 (Battery A) load description is shown on FSAR Table 8.3-6 and the Division 2 (Battery B) load description is shown on FSAR Table 8.3-7. The proposed change to the technical specifications results from increasing the allowable load (as shown in the FSAR) on the UPS from the present 50 amps to 150 amps on Battery A and from the present 115 amps to 150 amps on Battery B. The increase to 150 amps reflects a change to allow the maximum load rating for the UPS to be utilized.

At the first refueling outage the loads on Batteries A and B will be increased by a total of approximately 40 amps for each battery to allow for planned design modifications. These new loads will be added to the UPS which have adequate capacity, with margin, to accept the new loads. These planned design modifications include:

(1) Changing the electrical supply for Reactor Protection System Logic Circuits and Radiation and Neutron Monitoring Circuits from the Reactor Protection bus to Class 1E UPS.

(2) Changing Drywell Atmosphere Temperature Monitoring Circuits from Class 1E ESF to Class 1E UPS.

(3) Providing annunciation for UPS inverters in the control room.

(4) Changing RCIC flow control loop from ESF Division 1 to UPS.

The load increase on the Division 1 and 2 batteries has been evaluated to ensure that battery capacity has not been exceeded considering battery aging and temperature correction factors. Applicable provisions of IEEE 485-1978 and IEEE 450-1980 were considered in developing this proposed change to the technical specifications. FSAR Tables 8.3-6 and 8.3-7 will be updated to reflect the new load requirements after approval of this technical specification change request.

SIGNIFICANT HAZARDS CONSIDERATION:

The proposed change will increase the Division 1 and 2 battery load capacity testing requirements to reflect new loads to be added at the first refueling outage and to allow new loads to be added in the future. The new load profiles are within the capacity of the present batteries and are needed to allow implementation of scheduled plant design changes.

The proposed change does not involve a significant hazards consideration because operation of Grand Gulf Unit 1 in accordance with this change would not:

(1) involve a significant increase in the probability or consequences of an accident previously evaluated. This change merely increases the Division 1 and 2 battery capacity load tests, while ensuring that the new loads do not exceed battery capacity. Therefore, this change cannot increase the probability or consequences of an accident.

(2) create the possibility of a new or different kind of accident from any previously analyzed. It has been determined that a new or different kind of accident will not be possible due to this change. The load on the batteries for Division 1 and 2 will remain below rated capacity with this change thus ensuring that the emergency loads will receive necessary DC power for the assumed time frames in the accident analyses.

(3) involve a significant reduction in a margin of safety. The increased loads on the Division 1 and 2 batteries are within the battery capacities, therefore, all required loads will be supplied DC power for analyzed time frames and no margin of safety is reduced.

d. At least once per 18 months, during shutdown, by verifying that either:

1. The battery capacity is adequate to supply and maintain in OPERABLE status all of the actual emergency loads for 4 hours for Divisions 1 and 2 and 2 hours for Division 3 when the battery is subjected to a battery service test, or

2. The battery capacity is adequate to supply a dummy load of the following profile, which is verified to be greater than the actual emergency load, while maintaining the battery terminal voltage greater than or equal to 105 volts.

a) Division 1

≥950 amperes for the first 60 seconds
≥228 ~~≥120~~ amperes for the next 119 minutes
≥399 ~~≥306~~ amperes for the next 60 seconds
≥228 ~~≥120~~ amperes for the next 118 minutes
≥416 amperes for the last 60 seconds

b) Division 2

≥462 ~~≥427~~ amperes for the first 60 seconds
≥221 ~~≥106~~ amperes for the next 119 minutes
≥392 ~~≥357~~ amperes for the next 60 seconds
≥221 ~~≥106~~ amperes for the next 118 minutes
≥278 ~~≥243~~ amperes for the last 60 seconds

c) Division 3

≥76 amperes for the first 60 seconds
≥16 amperes for the next 59 minutes
≥18 amperes for the last 60 minutes

e. At least once per 60 months during shutdown by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Once per 60 month interval, this performance discharge test may be performed in lieu of the battery service test.

f. Annual performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

SSW ANALYSIS SUMMARY

Regulatory Guide 1.27 requires that the Ultimate Heat Sink (UHS) be capable of providing sufficient cooling water to dissipate residual heat (e.g. RHR, High Density Spent Fuel Storage Rack decay heat, etc.) after a design basis accident (DBA) for a period of thirty (30) days without inventory replenishment. To assure this requirement is satisfied, a Standby Service Water (SSW) system analysis has been performed to verify the capability of the SSW UHS to dissipate the residual heat associated with a DBA in Unit 1 with Unit 2 not in operation.

The total evaporative and drift losses from the UHS cooling towers for the 30 day period following the DBA in Unit 1 with Unit 2 not in operation has also been determined to verify the 30 day inventory requirement.

The system analysis is based on the following assumptions:

1) Unit 1 is experiencing a loss of coolant accident (LOCA) coincident with a loss of offsite power (LOP),

2) Unit 2 is not operational,

3) Worst single active failure occurs. The worst active failure for this analysis is the loss of one of the standby diesel generators which removes one of the SSW loops from operation (Standby diesel generator A is assumed to fail),

4) Makeup to the SSW system is not available from normal means of supply (Transfer between basin A and basin B is possible), and

5) Worst 30 day meteorology (per Regulatory Guide 1.27).

These assumptions will result in the greatest heat rejection rate for the UHS during the most severe meteorology following the DBA.

Analysis results indicate the total water losses from the SSW UHS cooling towers will be 7,784,971 gallons for the 30 day post LOCA period. The total usable basin B volume from elevation 84'-6" to elevation 130'-3" is 6,638,508 gallons. The total basin A volume from elevation 84'-6" to elevation 130'-3" is also 6,638,508 gallons. The total basin A usable volume, however, is limited to the volume above elevation 105'-0" (the inlet elevation of the basin siphon line) when utilizing the basin siphon. Therefore, the total basin A volume is reduced by 2,974,632 gallons (i.e. basin A volume from elevation 84'-6" to elevation 105"-0") so that the total usable basin A volume is 3,663,876 gallons. Thus, the total combined basin usable volume is 10,302,384 gallons. With the total water loss, evaporative and drift, calculated to be 7,784,971 gallons, there will be 2,517,413 gallons remaining at the end of the 30 day period.

Therefore, the performance analysis verifies the 30 day post LOCA inventory requirement. The analysis also verifies the capability of the UHS to dissipate the residual heat since that even with a gradual depleting basin water inventory due to evaporation and drift, the maximum cold water temperature of 87.0°F will still not exceed the design temperature of 90°F.