Attachment to
AECM-83/0180

PROPOSED CHANGE TO THE OPERATING LICENSE NPF-13
PCOL-83/03

Mississippi Power & Light (MP&L) requests that the operating license for Grand Gulf Nuclear Station (GGNS)(NPF-13) be amended as detailed below. These proposed changes, as discussed below, are provided for Nuclear Regulatory Commission (NRC) review and approval per 10 CFR 50.90.

A. SURVEILLANCE PROCEDURE REVIEW - PACKAGE NO. 1, (ITEMS 1 THROUGH 32).

1. (GGNS-204)

SUBJECT:

Technical Specification 3.6.4, Table 3.6.4-1, Page 3/4 6-44

DISCUSSION:

Page 3/4 6-44 of table 3.6.4-1, Containment and Drywell Isolation Valves is subheaded "Drywell (continued)." This is incorrect. The correct subheading should be "Containment (continued)."

JUSTIFICATION:

Page 3/4 6-44 is a continuation of Table 3.6.4-1, Section 4.a, Containment Test Connections which begins on page 3/4 6-42. Penetrations 67-92 covered on this page are containment penetrations as verified by Table 6.2-44 of the FSAR.

2. (GGNS-238)

SUBJECT:

Technical Specification 3.6.4, Table 3.6.4-1, page 3/4 6-30.

DISCUSSION:

Technical Specification Table 3.6.4-1, page 3/4 6-30, Containment and Drywell Isolation Valves lists the number for the RHR "C" Test Line to Suppression Pool as E12-F021B-B. This is incorrect. The correct number Should be E12-F021-B.

JUSTIFICATION:

FSAR Table 6.2-44 lists the RHR "C" Pump Test Line to Suppression Pool as E12-F021-B. This can also be confirmed on FSAR Figure 5.4-16.

3. (GGNS-234)

SUBJECT:

Technical Specification Bases 3.0.3, page B 3/4 0-1.

DISCUSSION:

Bases Section 3.0.3 cites as example "... Specification 3.6.6.1 requires two primary containment hydrogen recombiner systems to be OPERABLE...". This reference is incorrect. The correct reference should be "Specification 3.6.7.1."

JUSTIFICATION:

Technical Specification section 3.6.7.1 pertains to the Containment and Drywell Hydrogen Recombiner Systems and contains the requirements referenced in Bases Section 3.0.3.

4. (GGNS-235)

SUBJECT:

Technical Specification Bases Figure B 3/4 3-1, page B 3/4 3-7.

DISCUSSION:

Bases Figure B 3/4 3-1 indicates MSIV closure at low, low level 2(-41.6 inches) and also at low, low, low level 1(-150.3 inches). The level 2 trip is incorrect and reference to MSIV closure should be deleted from the figure for level 2.

JUSTIFICATION:

Technical Specification Table 3.3.2-2, Page 3/4 3-15 and FSAR Table 6.2-44 list MSIV trip setpoint to be low, low, low level 1(-150.3 inches) as the correct trip setpoint.

5. (GGNS-51)

SUBJECT:

Technical Specification 3/4.3.3, Table 3.3.3-2, page 3/4 3-28.

DISCUSSION:

Table 3.3.3-2, ECCS Actuation Instrumentation Setpoints, Section C.1.a, lists, the reactor water level-low, low level 2 trip setpoint for the HPCS system as greater than or equal to -41.6 inches and the allowable value as less than or equal to -43.8 inches. The allowable value is incorrect. The allowable value should read greater than or equal to -43.8 inches.

JUSTIFICATION:

The entire traversing in-core probe system mechanism is within the Mark III Containment Building, and the only containment penetrations are electrical. These isolation valves are therefore not required and are, in fact, not part of the GGNS design.

8. (GGNS-79)

SUBJECT:

Technical Specification Bases 3/4.7.2, page B 3/4 7-1

DISCUSSION:

Section 3/4.7.2 Control Room Emergency Filtration System, first paragraph, gives the bases for the operability requirement of Technical Specification 3/4.7.2. The second paragraph deals with surveillance requirements for the RCIC system. The second paragraph should be deleted.

JUSTIFICATION:

The second paragraph is identical to the last paragraph on page B 3/4 7-1 and does not apply to the control room emergency filtration system.

9. (GGNS-236, 300)

SUBJECT:

Technical Specification Table 3.6.4-1, Section 1.a, page 3/4 6-29, Section 3.a, page 3/4 6-37 and Section 4.a, page 3/4 6-42.

DISCUSSION:

Technical Specification Table 3.6.4-1, Section 1.a incorrectly lists valve number E12-F023B as a "RCIC and RHR to Head Spray" valve. The words "RCIC and" should be deleted from the phrase to reflect the GGNS design.

Technical Specification Table 3.6.4-1, Section 3.a also incorrectly lists valves E51-F066 and E51-F344 as "RCIC and RHR to Head Spray" valves. Also, the system designator E51 of valve number E51-F344 is incorrect. This is an E12 (RHR) valve and should be designated as E12-F344.

Technical Specification Table 3.6.4-1, Section 4.a contains incorrect numbers for the RHR to Head Spray Test connection isolation valves. These are listed as E51-F061 and E51-F342. Their correct numbers should be E12-F061 and E12-F342.

JUSTIFICATION:

GGNS "as built" design drawings and documents along with FSAR Figure 5.4-16 have been reviewed to verify the changes listed above.

JUSTIFICATION:

Basis 3/4.3.3 states that "Operation with a trip setpoint less conservative than its Trip Setpoint but within its specified allowable value is acceptable on the basis that the difference between each trip setpoint and the allowable value is equal to or less than the drift allowance assumed for each trip in the safety analysis." An allowable value of less than or equal to -43.8 inches would allow the trip setpoint to be below the -43.8 level which is non conservative. The correct allowable value is greater than or equal to -43.8 inches which would be the lower bound of the drift allowance.

6. (GGNS-319)

SUBJECT:

Technical Specification 3/4.3.2, Table 3.3.2-1, Section 3.c and 3.d, page 3/4 3-11, 14.

DISCUSSION:

Table 3.3.2-1, Items 3.c and 3.d incorrectly lists the isolation which occurs on high radiation in the Auxiliary Building Fuel Handling Area and Fuel Pool Sweep Exhaust. A high radiation condition in these areas causes the isolation of the Auxiliary Building ventilation isolation dampers, starts the Standby Gas Treatment System and starts the D/P recorder in fast speed. In Table 3.3.2-1, these signals are incorrectly recorded as causing group 6 isolation valve closures. References to group six closure should be deleted.

JUSTIFICATION:

FSAR sections 9.4.2.2., 9.4.2.3 and 9.4.2.5 discuss the requirements to prevent an uncontrolled release of radioactivity from this radiation - High High Signal. Group 6 isolation valve closures are not part of the requirements. In fact, no specified isolation valve group is closed by this signal. Only the Auxiliary Building ventilation isolation dampers are closed by this signal which is specified in the proposed new note.

7. (GGNS-50, 94)

SUBJECT:

Technical Specification 4.6.4.4, page 3/4 6-28

DISCUSSION:

Surveillance Requirement 4.6.4.4 requires that each traversing in-core probe system explosive isolation valve be demonstrated operable. These explosive isolation valves are not used in GGNS-1; therefore, the section should be deleted.

10. (GGNS-271)

SUBJECT:

Technical Specification 3.1.3.2, Section b, Page 3/4 1-7.

DISCUSSION:

Section b of Technical Specification 3.1.3.2, Control Rod Maximum Scram Insertion Times, refers to action statement a.l.a. This reference is incorrect. The correct reference should be a.l.

JUSTIFICATION:

Action Statement a.l.a as referenced in section b does not exist. Section b should reference action statement a.l.

11. (GGNS-274)

SUBJECT:

Technical Specification 3.3.7.11, Table 4.3.7.11-1, page 3/4 3-85

DISCUSSION:

Table 4.3.7.11-1 lists the Radioactive Liquid Effluent Monitoring Instrumentation Surveillance Requirements. The second item, flow rate measurement devices, is incorrectly numbered with a three (3) instead of a two (2). The three (3) should be deleted and replaced with a two (2).

JUSTIFICATION:

The second item was inadvertently numbered three (3) instead of two (2).

12. (GGNS-74)

SUBJECT:

Technical Specification 4.8.3.1.1 and 4.8.3.2.1, pages 3/4 8-16, 18.

DISCUSSION:

The subject surveillance requirements currently require at least the power distribution system divisions as indicated in Technical Specification 3.8.3.1 and 3.8.3.2 to be determined energized by verifying correct breaker alignment and voltage on the Busses/MCCs/panels.

Technical Specification 4.8.3.1.1 and 4.8.3.1.2 should be revised to read "... correct breaker alignment on the Busses/LCs/MCCs/panels and voltage on the Busses/LCs".

JUSTIFICATION:

The above mentioned changes reflect the types of motor control centers and panels employed in the Grand Gulf design. There is no instrumentation on MCCs or panels for voltage readings. The LCs which feed the MCs have installed instrumentation for voltage readings as do the busses which feed the LCs.

13. (GGNS-324)

SUBJECT:

Technical Specification 4.1.4.2, Section a.2, page 3/4 1-17. Technical Specification 3.3.6, Table 3.3.6-2, Section 1.a, page 3/4 3-52.

DISCUSSION:

Technical Specification Table 3.3.6-2, Control Rod Block Instrumentation Setpoints and Surveillance Requirements 4.1.4.2.a.2 incorrectly state the Rod Pattern Control System (RPCS) low power set point as less than or equal to 20% of Rated Thermal Power. According to the appropriate General Electric design specification, the low power setpoint and allowable value should be 20% + 15%,-0% of Rated Thermal Power.

JUSTIFICATION:

Bases 3/4.1.4 states that when Thermal Power greater than 20% of Rated Thermal Power, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Therefore, the RPCS is required to be operable when Thermal Power is less than or equal to 20% of Rated Thermal Power. In order to assure this, the low power setpoint must be at least 20% of Rated Thermal Power.

14. (GGNS-257)

SUBJECT:

Technical Specification 3.8.4, Table 3.8.4.1-1, page 3/4 8-21

DISCUSSION:

The proposed change involves revising the Trip Setpoint for the 6.9 KV circuit breakers which protect the Reactor Recirculation Pump Motors. Since the instantaneous overcurrent protection devices trip the Reactor Recirculation Pump Motor breakers when switching from low to high speed but do not trip during a direct high speed start, the trip setpoint specified in the Technical Specifications should be revised slightly to reflect the locked rotor current rise due to the residual voltage.

Breaker Number 252-1003-C is misnumbered. The correct number is 252-1103-C.

JUSTIFICATION:

An engineering evaluation was performed to insure that the higher current values are appropriate and that these devices continue to function as designed. In order to provide better protection for an inside containment fault, the overcurrent unit time dial setting will be decreased from 9 to 8 seconds. FSAR Figure 040.5-1 shows that an increase from 6400/40A to 7200/45A will not adversely affect the penetration or cable.

15. (GGNS-297, 320)

SUBJECT:

Technical Specification 4.8.2.1, Section a.1, page 3/4 8-11 and Table 4.8.2.1-1, page 3/4 8-13.

DISCUSSION:

Table 4.8.2.1-1 gives the limits for specific gravity for the Division 1, Division 2, and Division 3, 125 volt batteries. These limits, however, do not reflect the manufacturer's recommended value for specific gravity. Table 4.8.2.1-1 should be amended to reflect manufacturer's specifications.

JUSTIFICATION:

According to vendor specifications, the manufacturer's full charge specific gravity for the 125 volt batteries installed at GGNS is 1.210. Therefore, the Battery Surveillance Requirements given in Table 4.8.2.1-1 should be changed to reflect this value. The changes indicated on the markup page 3/4 8-13 conform to the normal limits specified in Bases Section 3/4.8.2.

16. (GGNS-302)

SUBJECT:

Technical Specification 3.3.2, Table 3.3.2-3, page 3/4 3-18 .

DISCUSSION:

Table 3.3.2-3, Section 1.c, Response Time for Containment and Drywell Ventilation Exhaust Radiation - High, gives a response time for MSIV closure. These instruments do not cause isolation of the MSIVs. This response time requirement should be deleted from this table.

JUSTIFICATION:

FSAR Section 7.3.1.1.2.4.1.7 discusses the Containment and Drywell Ventilation Exhaust Radiation Monitoring Instrumentation and Controls. Section 7.3.1.1.2.4.1.7.4 states that a trip causes isolation of all containment and drywell ventilation penetrations. FSAR figure 7.6-1 confirms the fact that no MSIV isolations occur.

17. (GGNS-352, 356)

SUBJECT:

Technical Specifications 3/4.3.7.11, Table 4.3.7.11-1, page 3/4 3-85; 3/4.11.1, Table 4.11.1.1.1-1, page 3/4 11-2; and Definitions Table 1.1, page 1-9.

DISCUSSION:

Table 4.3.7.11-1, Radiological Liquid Effluent Monitoring Instrumentation Surveillance Requirements and Table 4.11.1.1.1-1, Radioactive Liquid Waste Sampling and Analysis Program, use "P" as a surveillance frequency notation. The letter "P," as used in these tables, is intended to designate "Completed prior to each release". This definition, however, is not given in these tables nor in Table 1.1, page 1-9 with the other surveillance frequency notations.

The following should be added to Table 1.1, page 1-9:

Notation	Frequency
P	Completed prior to each release

JUSTIFICATION:

The letter "P" is used in such a manner in the Standard Radiological Environmental Technical Specification and is also defined as such in the Grand Gulf Nuclear Station Operations Manual Administrative Procedure 01-S-06-12 Rev. 5, Attachment 1.

18. (GGNS-362)

SUBJECT:

Technical Specification 3/4.6.6, Table 3.6.6.2-1, page 3/4 6-48.

DISCUSSION:

Technical Specification Table 3.6.6.2-1, Secondary Containment Ventilation System Automatic Isolation Dampers/ Valves, incorrectly gives the number of the Fuel Handling Area Ventilation Supply Dampers as (Q1T42 F0011) and (Q1T42 F0012). These dampers should be correctly listed as (Q1T42 F011) and (Q1T42 F012) respectively.

JUSTIFICATION:

FSAR section 9.4.2 describes the Fuel Handling Area Ventilation System Figure 9.4-2 in the FSAR shows the dampers in question and gives their correct numbers.

19. (GGNS-102, 384)

SUBJECT:

Technical Specification 3/4.3.2.12, Tables 3.3.7.12-1 and 4.3.7.12-1 on pages 3/4 3-88, 3/4 3-90, 3/4 3-91 and 3/4 3-92.

DISCUSSION:

Items 3.a in Table 3.3.7.12-1 and Item 3.a in Table 4.3.7.12-1 are designated as: "Noble Gas Activity Monitor Providing Alarm and Automatic Termination of Release." This designation should be changed to eliminate the reference to automatic termination of release. The action statement for item 3.a in Table 3.3.7.12-1 should be changed to action 121. Action statement 125 should be deleted from page 3-91 and statement 126 should accordingly be renumbered 125. Finally, the action statement for Item 6.a in Table 3.3.7.12-1 should be revised to 125.

JUSTIFICATION:

Automatic termination of radioactive effluent releases is provided by the containment and drywell ventilation exhaust radiation monitors which are listed as item 7 in Technical Specification Table 3.3.7.1-1. These monitors automatically isolate the containment on high radiation. The noble gas monitors, identified as item 3.a in Tables 3.3.7.12-1 and 4.3.7.12-1, do not provide any automatic isolation function. Therefore reference to automatic termination of release should be deleted from their designation. Since the automatic isolation function is provided by separate monitors, the requirements of action statement 125, i.e., suspension of release of radioactive effluent from the containment ventilation system, are not warranted. The same actions should be required as a result of inoperability of these monitors as the actions required due to inoperability of other building ventilation noble gas activity monitors. The action statement for item 3.a in Table 3.3.7.12-1 should be changed to 121. Since item 3.a is the only item which references action statement 125, action statement 125 should be deleted on page 3/4 3-91 and action statement 126 should accordingly be renumbered Action 125. This necessitates changing the action statement for item 6.a in Table 3.3.7.12-1 from 126 to 125.

20. (GGNS-173, 237)

SUBJECT:

Technical Specification Table 3.3.2-2, page 3/4 3-15.

DISCUSSION:

The main steam line isolation trip setpoint and allowable value for main steam line radiation are currently specified as less than or equal to 1.5 times full power background and 3.0 times full power background respectively in Table 3.3.2-2, Item 2.b. These setpoint and allowable values are inconsistent with the correct setpoints which are identified elsewhere in the technical specification. The correct trip setpoint for item 2.b in table 3.3.2-2 should be less than or equal to 3.0 times full power background and the allowable value for item 2.b should be less than or equal to 3.6 times full power background.

JUSTIFICATION:

Subsection 7.3.1.1.2.4.1.2.3 of the FSAR states that the main steam line radiation monitors provide dual functions in actuating the reactor protection system and closing the main steam isolation valves at the same setpoint. Technical Specification Table 2.2.1-1 item 7 requires that the main steam line radiation monitors should trip the reactor protection system at less than or equal to 3.0 times full power background with an allowable value of less than or equal to 3.6 times full power background. This is the correct trip setpoint and allowable value based upon NSSS vendor documentation.

21. (GGNS-256)

SUBJECT:

Technical Specification 3.3.7, Table 3.3.7.1-1, page 3/4 3-56.

DISCUSSION:

The range for items 1, 2, and 4 of Technical Specification Table 3.3.7.1-1 is incorrectly listed as 1 to 10^6 counts per minute. The correct range for these instruments should be listed in this table as 10 to 10^6 counts per minute.

JUSTIFICATION:

The correct range for these instruments, (the Component Cooling Water radiation monitor, the Standby Service Water System radiation monitor, and the Offgas Post Treatment radiation monitor), is provided in Table 11.5-1 of the FSAR. The correct range is 10 to 10^6 counts per minute.

22. (GGNS-354)

SUBJECT:

Technical Specification 4.8.1.1.2.d.16, page 3/4 8-7.

DISCUSSION:

In Item f of Surveillance Requirement 4.8.1.1.2.d.16, the word "generator" should be replaced with the word "engine" such that Item f reads "engine bearing temperature high (11 and 12 only). In Item j, the parenthetical expression "(13 only)" should be appended to the trip listing such that Item j reads "low lube oil pressure (13 only)".

JUSTIFICATION:

Bearing Temperature Monitoring

A review of the installed electrical drawings shows that the lockout features listed in section d.16 are reversed. High engine bearing temperature for diesel 11 and 12 prevents these diesel generators from starting or trips the generators during operation if the LOCA and LOP signals are absent. A high generator bearing temperature signal will produce an alarm on the local panel for diesel 11 and 12. This design provides superior protection for the diesel engines. Engine bearing temperature is more critical with respect to maintaining the ability of the diesel generators to fulfill their specified safety function. The diesel generators 11 and 12 should be tripped, during tests, on high engine bearing temperature.

Low Lube Oil Pressure Monitoring

Surveillance requirement 4.8.1.1.2.d.16 requires verification of diesel generator lockout features which prevent the diesel generator from starting and which trip the diesel generator when it is operating without the presence of the LOCA or LOP signals. Surveillance requirement 4.8.1.1.2.d.8 requires verification of trips which prevent the diesel generators from starting and which trip the diesel generators when they are operating even with the presence of the LOCA or LOP signals. FSAR subsection 8.3.1.1.4.1.f states that a low lube oil pressure will prevent starting or trip operation of diesel generators 11 and 12 even if the LOCA and LOP signals are present. Therefore, the low lube oil pressure lockout feature in Item j of surveillance requirement 4.8.1.1.2.d.16 applies only to diesel generator 13.

23. (GGNS-158, 292, 17)

SUBJECT:

Technical Specification Table 3.8.4.2-1, pages 3/4 8-40 through 3/4 8-45.

DISCUSSION:

Technical Specification Table 3.8.4.2-1 lists the safety related motor operated valves which are furnished with thermal overload protection. This table requires revision to include additional valves and certain changes to accurately reflect the plant design.

JUSTIFICATION:

Regulatory Guide 1.106 specifies requirements for design of safety related motor operated valve thermal overload protection devices. This regulatory guide controls application of continuous bypasses, bypasses under accident conditions and identifies when bypasses are not required for thermal overload protection devices. MP&L committed to implement this regulatory guide in Appendix 3A of the FSAR. The changes noted on the attached pages reflect additional valves which fulfill a safety function and should be added to the table or changes which are necessary to reflect the correct regulatory requirements and accordingly, the as-installed arrangement.

24. (GGNS-501)

SUBJECT:

Technical Specification Table 3.3.7.5-1, page 3/4 3-71.

DISCUSSION:

A portion of the text for Action 80, part b, has been inadvertently omitted from the technical specifications. Action 80, part b, should be revised to read as follows:

"b. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3.7.5-1, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours."

JUSTIFICATION:

The text currently in the technical specifications for Action 80, part b is unclear. The revised text is taken from the corresponding section of NUREG 0123, Standard Technical Specifications for General Electric Boiling Water Reactors.

25. (GGNS-452)

SUBJECT:

Technical Specification 4.9.12, page 3/4 9-18

DISCUSSION:

Technical Specification 4.9.12 specifies that the surveillance requirements for the Horizontal Fuel Transfer System (HFTS) should be completed within 4 hours prior to operation of HFTS and at least once per 12 hours thereafter. This requirement should be changed to specify that the surveillance requirements should be completed within 24 hours prior to operation of HFTS and at least once per 7 days thereafter.

JUSTIFICATION:

The surveillance requirements for the HFTS presently included in the technical specifications are appropriate for an Inclined Fuel Transfer System (IFTS). An IFTS includes interlocks to prevent draining of the upper containment fuel pool and to prevent overexposure of personnel due to inadvertent opening of certain critical doors. The HFTS installed at

Grand Gulf Nuclear Station does not include these interlocks since the system cannot result in draining the upper containment pools and personnel access to high radiation areas is blocked with concrete shield walls. The surveillance requirements for the HFTS should be made consistent with the surveillance requirements for the refueling interlocks specified in surveillance requirements 4.9.1.2. These requirements specify that the refueling system interlocks should be verified to be operable within 24 hours prior to commencing core alterations and at least once per 7 days thereafter.

26. (GGNS-416, 417, 433)

SUBJECT:

Technical Specification 3.9.1.b, page 3/4 9-1.

DISCUSSION:

The wording for item 3.9.1.b.3 should be revised to read as follow:

"3. Refuel platform main hoist fuel-loaded"

Items 3.9.1.b.4 and 3.9.1.b.5 should be deleted.

JUSTIFICATION:

The four interlock conditions monitored by the Refueling Interlock System are: refueling platform positioned near or over the core, refueling platform main hoist fuel-loaded, reactor mode switch position, and control rod position.

The fuel grapple not full up interlock has been deleted from the GGNS design. This interlock was a backup interlock to assure that no single failure could permit the refueling platform to be positioned over the core, a fuel assembly lifted, and a control rod to be inadvertently withdrawn. The need for this interlock was eliminated when redundant circuits were installed to sense the positioning of the refueling platform over the core and when the refueling platform main hoist is loaded with fuel. Since the fuel grapple position interlock has been deleted in the GGNS Design, Technical Specification 3.9.1.b.4 should be deleted.

The other two hoists on the refueling platform are furnished with load sensors which specifically prohibit these hoists from lifting fuel. The only hoist which needs to have a hoist fuel-loaded interlock is the main hoist. Therefore, Technical Specification 3.9.1.b.3 should be revised to reflect applicability of the fuel loaded interlock to only the main hoist.

No interlocks are provided between the source range monitor count-rates and the refueling equipment. Technical Specification 3.9.2 controls operability requirements for the source range monitors when the plant is

JUSTIFICATION:

The trip functions for items C.1.d and C.1.e in Technical Specification Table 3.3.3.1-1 are only required to be operable for operational conditions 4 and 5 when the HPCS is aligned to take suction from the condensate storage tank and the suppression pool is operable. The existing asterisk note in Table 3.3.3-1 adequately states that these channels need not be operable in operational condition 4 and 5 unless the specified conditions are satisfied. Equally, surveillance requirements for items C.1.d and C.1.e need not be imposed in operational conditions 4 and 5 by Technical Specification Table 4.3.3.1-1 unless the same conditions are satisfied.

The asterisk (*) note for Technical Specification Table 4.3.3.1-1 is unclear and is inconsistent with the asterisk note for Table 3.3.3.1-1. The surveillance requirements for items C.1.a, C.1.c, C.1.d, C.1.e, and C.1.f in operational conditions 4 and 5 should be performed whenever the HPCS system is required to be operable by specifications 3.5.2 or 3.5.3. The modifications to the asterisk note clarify the requirements for performing surveillance of the HPCS actuation instrumentation and make these surveillance requirements consistent with the operability requirements specified in Table 3.3.3.1-1.

29. (GGNS-99)

SUBJECT:

Technical Specification Table 3.3.2-1, page 3/4 3-14.

DISCUSSION:

Note i in Technical Specification Table 3.3.2-1 indicates that the Standby Liquid Control System actuation logic closes only the Reactor Water Cleanup (RWCU) system inlet outboard valve G33-F004. This signal also isolates RWCU valves G33-F001 and G33-F251 and note i should be revised to reflect that these valves are also isolated.

JUSTIFICATION:

Table 6.2-44 in the FSAR shows that for penetration number 87, signal Y, which corresponds to the Standby Liquid Control System actuation isolation signal, isolates valves G33-F004, G33-F001 and G33-F251. Therefore, note i in Technical Specification Table 3.3.2-1 should be revised to include these additional valves.

30. (GGNS-470)

SUBJECT:

Technical Specification Table 3.3.7.3-1, page 3/4 3-64 and Table 4.3.7.3-1, page 3/4/3-65.

in operational condition 5. Technical Specification item 3.9.1.b.5 should, therefore, be deleted. Therefore, Technical Specification 3.9.1.b should be revised as discussed above to reflect the interlocks which are present in the installed design.

27. (GGNS-303)

SUBJECT:

Technical Specification Tables 3.3.2-1, page 3/4 3-10, 3.3.2-2, page 3/4 3-15, 3.3.2-3, page 3/4 3-18, and 4.3.2.1-1, page 3/4 3-20.

DISCUSSION:

Item 1.c in each of the subject technical specification tables is labeled as Containment and Drywell Ventilation Exhaust Radiation - High. The correct designation for this primary containment isolation function should be Containment and Drywell Ventilation Exhaust Radiation - High-High.

JUSTIFICATION:

Each channel for the containment and drywell ventilation exhaust radiation monitors has three trips. The downscale trip indicates instrument trouble. The middle trip function initiates an alarm in the control room. The upscale trip produces an alarm and is capable of isolating the drywell and containment ventilation penetrations. As stated in FSAR Subsection 7.3.1.1.2.4.1.7.2 through 7.3.1.1.2.4.1.7.4, four channels are provided with two channels being powered from each RPS bus. Two upscale or high-high trips, two downscale or instrument inoperative trips, or one high-high and one instrument inoperative trips on either set of two channels will provide a trip signal for isolation of all containment and drywell ventilation penetrations.

28. (GGNS-2[illegible]9)

SUBJECT:

Technical Specification Table 4.3.3.1-1 page 3/4 3-33.

DISCUSSION:

The asterisk (*) notes for modes 4 and 5 under Applicable Operational Conditions for item C in Technical Specification Tables 3.3.3-1 and 4.3.3.1-1 are not consistent. The asterisk note in Table 4.3.3.1-1 should be revised to read as follows:

> "* Applicable when the system is required to be operable per specification 3.5.2 or 3.5.3."

DISCUSSION:

Technical Specification Tables 3.3.7.3-1 and 4.3.7.3-1 define requirements for the meteorological monitoring instrumentation. Both tables define requirements for the air temperature instruments at elevation 162 feet. Item c in tables 3.3.7.3-1 and 4.3.7.3-1 should be revised to delete reference to the temperature instruments at elevation 162 feet.

JUSTIFICATION:

Item d in Technical Specification Tables 3.3.7.3-1 and 4.3.7.3-1 defines requirements for the instruments which measure air temperature difference between elevations 33 feet and 162 feet. The temperature sensor at elevation 162 feet only provides input to this instrument. No separate indication is provided for the temperature sensor at elevation 162 as opposed to the separate indication which is provided for the temperature sensor at elevation 33 feet. The operability requirements and surveillance requirements specified for item d in tables 3.3.7.3-1 and 4.3.7.3-1 provide for operability and surveillance for the temperature sensor at elevation 162 feet. Data available in the control room from the meteorological instruments are discussed in FSAR subsection 2.3.3.2.2.2.

31. (GGNS-457)

SUBJECT:

Technical Specification Tables 4.3.6-1, items 2.a and 2.c, page 3/4 3-53.

DISCUSSION:

The channel calibrations for the control rod block instrumentation Average Power Range Monitors (APRMs) should be revised as follows:

	2 APRM	Channel Calibration
a.	Flow Biased Neutron Flux Upscale	$W^{(f)(g)}$, SA
c.	Downscale	$W^{(h)}$, SA

The following notes should be added to Technical Specification Table 4.3.6-1.

f. 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 is greater than or equal to 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.

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

h. This calibration shall consist of verifying the trip setpoint only.

JUSTIFICATION:

The channel calibration requirements currently specified in Technical Specification Table 4.3.6-1, item 2.a, for the APRM initiated control rod blocks, do not agree with the channel calibration requirements currently specified in Technical Specification Table 4.3.1.1-1, item 2.b, for the APRM reactor protection system function. Table 4.3.6-1 presently requires quarterly calibration of the APRM initiated control rod block for flow biased neutron flux upscale. Table 4.3.1.1-1 requires semi-annual calibration of the APRM reactor protection system function with weekly verification that the adjustment of the APRM channels are valid. The APRM functions associated with the reactor protection system are more significant with respect to plant safety than the APRM initiated control rod blocks. Therefore, the channel calibration frequency requirements for APRM initiated control rod blocks as specified in item 2a of Table 4.3.6-1 should be revised to conform with the requirements in Table 4.3.1.1-1 for item 2.b, which defines requirements for channel calibration frequency of the APRM reactor protection system function. The frequency of channel calibration of the APRM initiated control rod blocks should not be greater than the frequency of channel calibration for the APRM reactor protection system functions.

The downscale rod block identified as item 2.c in Table 4.3.6-1 has no corresponding reactor protection system surveillance requirement in Table 4.3.1.1-1. In order to be consistent with the upscale rod block listed as item a in Table 4.3.6-1, the channel calibration frequency should be revised to semi-annually with a requirement that this channel calibration is limited to verifying trip setpoint only.

This change is also necessary to support MP&L's ALARA program. Since full calibration would include entry into the containment, quarterly calibration of APRM initiated rod blocks would entail unnecessary exposure to operating personnel given that semi-annual calibration is acceptable for the APRM reactor protection system functions. Therefore, the channel calibration frequency for the APRM initiated rod blocks should be revised as shown above.

32. (GGNS-427)

SUBJECT:

Technical Specification Tables 3.3.7.2-1, page 3/4 3-61 and 4.3.7.2-1, page 3/4 3-62.

DISCUSSION:

Item 4 in Technical Specification Tables 3.3.7.2-1 and 4.3.7.2-1 should read:

"4. Vertical Seismic Trigger"

This instrument is not a recorder.

JUSTIFICATION:

The vertical seismic trigger identified as item 4 in Technical Specification Tables 3.3.7.2-1 and 4.3.7.2-1 is similar in function to the horizontal seismic trigger identified in item 5 of these tables. The word "recorder" should be deleted from item 4 in Tables 3.3.7.2-1 and 4.3.7.2-1.

B. MISCELLANEOUS TECHNICAL SPECIFICATION CHANGES, (ITEMS 33 THROUGH 36).

33. SUBJECT:

Technical Specification 6.5.2, page 6-9.

DISCUSSION:

Remove the asterisk and delete the footnote from Technical Specification 6.5.2.

JUSTIFICATION:

The advisor to the Assistant Vice-President, Nuclear Operations is presently limited in effectiveness by the provisions of Technical Specification 6.5.2, which make him a non-voting member of the Safety Review Committee (SRC). To make more effective use of the experience of the Advisor, remove the asterisk and reference footnote and allow the Advisor to be a voting member of the SRC.

34. SUBJECT:

Technical Specification 3.11.1.3. page 3/4 11-6.

DISCUSSION:

Technical Specification 3.11.1.3 refers to ODCM Figure 5.1.4.1 which is an error. The correct Figure is 5.1.3-1.

JUSTIFICATION:

Remove Figure number 5.1.4.1 since it does not exist and insert the correct ODCM Figure 5.1.3-1.

35. SUBJECT:

Technical Specification 6.5.2.10, page 6-12.

DISCUSSION:

Technical Specification 6.5.2.10 should be revised to modify the documentation procedure for SRC review activities.

JUSTIFICATION:

Technical Specification 6.5.2.10 should be revised to more accurately reflect required audit practices - that is, the Operational Quality Assurance manual requires the Manager of Quality Assurance to conduct audits of SRC written reports. This Technical Specification 6.5.2.10 should be revised to remove the redundant requirement of separate audit reports by the SRC and to more accurately reflect current administrative procedures consistent with this philosphy.

1. (GGNS-204)

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM AND VALVE NUMBER		PENETRATION NUMBER
Containment ~~Drywell~~ (Continued)		
~~RHR~~ "B" Test Line T/C	E12-F350	67(0)(c)
RHR "B" Test Line T/C	E12-F312	67(0)(c)
RHR "B" Test Line T/C	E12-F305	67(0)(c)
Refueling Water Transf. Pump Suction T/C	P11-F425	69(0)(c)
Refueling Water Transf. Pump Suction T/C	P11-F132	69(0)(c)
Inst. Air to ADS T/C	P53-F043	70(0)
Cont. Leak Rate T/C	M61-F010	82(I)
RWCU To Feedwater T/C	G33-F055	83(0)
Suppr. Pool Cleanup T/C	P60-F011	85(0)
Suppr. Pool Cleanup T/C	P60-F034	85(0)
RWCU Pump Suction T/C	G33-F002	87(0)
RWCU Pump Discharge T/C	G33-F061	88(0)
SSW T/C	P41-F163A	89(0)(c)
SSW T/C	P41-F163B	92(0)(c)
b. Drywell		
LPCI "A" T/C	E12-F056A	313(0)
LPCI "B" T/C	E12-F056B	314(0)
Intrument Air T/C	P53-F493	327(0)
SLCS T/C	C41-F026	328(0)
Service Air T/C	P52-F476	363(0)
Reactor Sample T/C	B33-F021	465(0)

2.(6GNS-238)

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)				
RHR Heat Exchanger "A" to LPCI	E12-F028A-A	20(I)(c)	5	78
RHR Heat Exchanger "A" to LPCI	E12-F037A-A	20(I)(c)	3	63
RHR Heat Exchanger "B" to LPCI	E12-F042B-B	21(I)(c)	5	22
RHR Heat Exchanger "B" to LPCI	E12-F028B-B	21(I)(c)	5	78
RHR Heat Exchanger "B" to LPCI	E12-F037B-B	21(I)(c)	3	63
RHR "A" Test Line to Supp. Pool	E12-F024A-A	23(O)(d)	5	93
RHR "A" Test Line to Supp. Pool	E12-F011A-A	23(O)(d)	5	27
RHR "A" Test Line to Supp. Pool	E12-F290A-A	23(O)(d)	6	8
RHR "C" Test Line to Supp. Pool	E12-F021-B ~~E12-F021B-B~~	24(O)(d)	5	101
HPCS Test Line	E22-F023-C	27(O)	6	60
RCIC Pump Suction	E51-F031-A	28(O)	4	38
RCIC Turbine Exhaust	E51-F077-A	29(O)(c)	9	18
LPCS Test Line	E21-F012-A	32(O)	5	101
Cont. Purge and Vent Air Supply	M41-F011	34(O)	7	4
Cont. Purge and Vent Air Supply	M41-F012	34(I)	7	4
Cont. Purge and and Vent Air Exh.	M41-F034	35(I)	7	4
Cont. Purge and and Vent Air Exh.	M41-F035	35(O)	7	4
Plant Service Water Return	P44-F070-B	36(I)	6	24
Plant Service Water Return	P44-F069-A	36(O)	6	24
Plant Service Water Supply	P44-F053-A	37(O)	6	24
Chilled Water Supply	P71-F150	38(O)	6	30
Chilled Water Return	P71-F148	39(O)	6	30

3.(GGNS-234)

3/4.0 APPLICABILITY

BASES

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4.

3.0.1 This specification states the applicability of each specification in terms of defined OPERATIONAL CONDITION or other specified applicability condition and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.3 This specification delineates the measures to be taken for those circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of the specification. For example, Specification 3.7.2 requires two control room emergency filtration subsystems to be OPERABLE and provides explicit ACTION requirements if one subsystem is inoperable. Under the requirements of Specification 3.0.3, if both of the required subsystems are inoperable, within one hour measures must be initiated to place the unit in at least STARTUP within the next 6 hours, in at least HOT SHUTDOWN within the following 6 hours and in at least COLD SHUTDOWN within the subsequent 24 hours. As a further example, Specification ~~3.6.6.1~~ 3.6.7.1 requires two primary containment hydrogen recombiner systems to be OPERABLE and provides explicit ACTION requirements if one recombiner system is inoperable. Under the requirements of Specification 3.0.3, if both of the required systems are inoperable, within one hour measures must be initiated to place the unit in at least STARTUP within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

3.0.4 This specification provides that entry into an OPERATIONAL CONDITION must be made with (a) the full complement of required systems, equipment or components OPERABLE and (b) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out of service provisions contained in the ACTION statements.

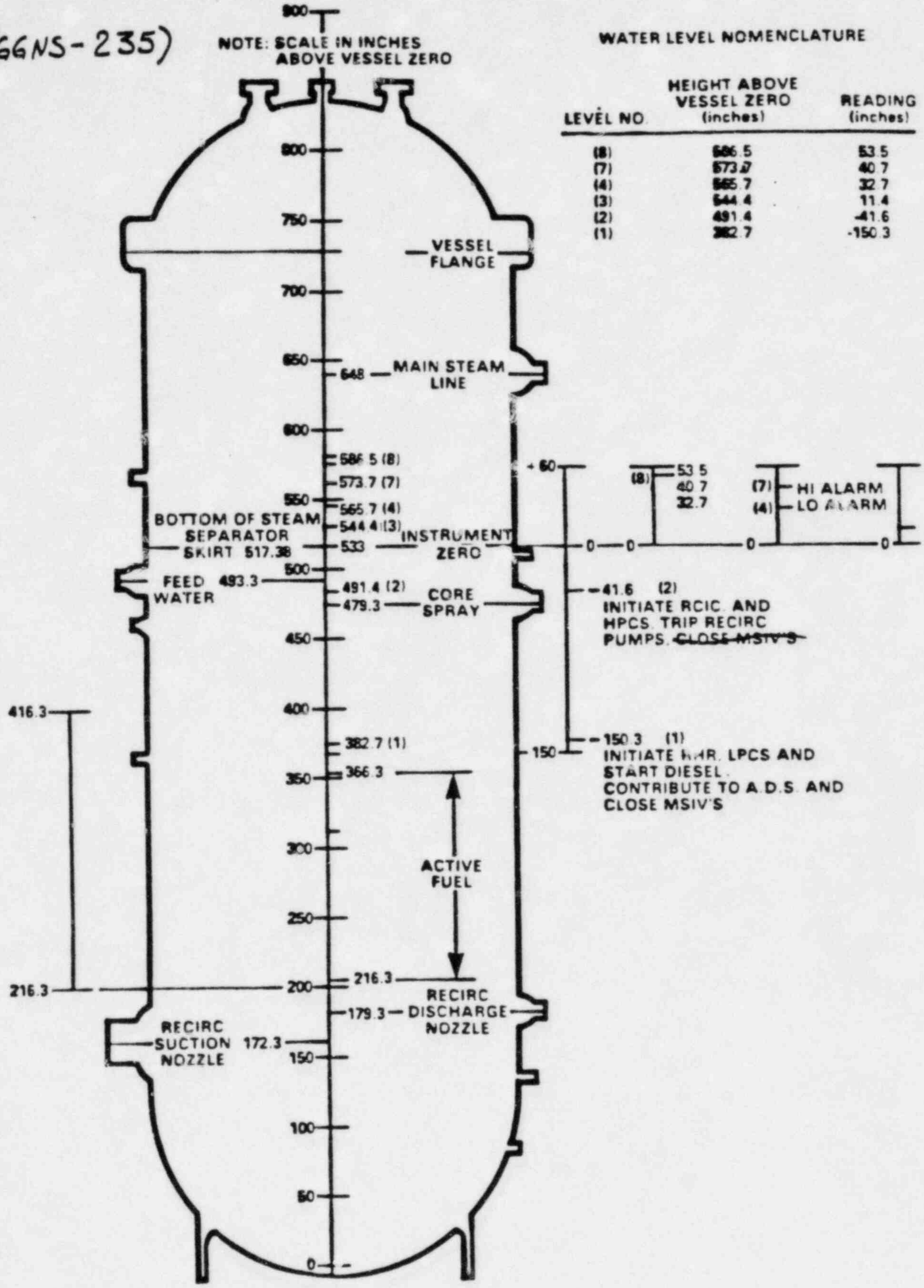
The intent of this provision is to ensure that unit operation is not initiated with either required equipment or systems inoperable or other limits being exceeded.

Exceptions to this provision have been provided for a limited number of specifications when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications.

4. (GGNS-235)

WATER LEVEL NOMENCLATURE

LEVEL NO.	HEIGHT ABOVE VESSEL ZERO (inches)	READING (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

5.(GGNS-51)

TABLE 3.3.3-2

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
A. DIVISION 1 TRIP SYSTEM		
1. RHR-A (LPCI MODE) AND LPCS SYSTEM		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -150.3 inches*	≥ -152.5 inches
b. Drywell Pressure - High	≤ 1.89 psig	≤ 1.94 psig
c. LPCI Pump A Start Time Delay Relay	≤ 5 seconds	≤ 5 seconds
d. Manual Initiation	NA	NA
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "A"		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -150.3 inches*	≥ -152.5 inches
b. Drywell Pressure - High	≤ 1.89 psig	≤ 1.94 psig
c. ADS Timer	≤ 115 seconds	≤ 117 seconds
d. Reactor Vessel Water Level-Low, Level 3	≥ 11.4 inches*	≥ 10.8 inches
e. LPCS Pump Discharge Pressure-High	≥ 145 psig, increasing	≥ 140 psig, increasing
f. LPCI Pump A Discharge Pressure-High	≥ 125 psig, increasing	≥ 122 psig, increasing
g. Manual Initiation	NA	NA
B. DIVISION 2 TRIP SYSTEM		
1. RHR B AND C (LPCI MODE)		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -150.3 inches*	≥ -152.5 inches
b. Drywell Pressure - High	≤ 1.89 psig	≤ 1.94 psig
c. LPCI Pump B Start Time Delay Relay	≤ 5 seconds**	≤ 5 seconds
d. Manual Initiation	NA	NA
2. AUTOMATIC DEPRESSURIZATION SYSTEM TRIP SYSTEM "B"		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -150.3 inches*	≥ -152.5 inches
b. Drywell Pressure - High	≤ 1.89 psig	≤ 1.94 psig
c. ADS Timer	≤ 115 seconds	≤ 117 seconds
d. Reactor Vessel Water Level-Low, Level 3	≥ 11.4 inches*	≥ 10.8 inches
e. LPCI Pump B and C Discharge Pressure-High	≥ 125 psig, increasing	≥ 122 psig, increasing
f. Manual Initiation	NA	NA
C. DIVISION 3 TRIP SYSTEM		
1. HPCS SYSTEM		
a. Reactor Vessel Water Level - Low Low, Level 2	≥ -41.6 inches*	≥ ~~≤~~ -43.8 inches
b. Drywell Pressure - High	≤ 1.89 psig	≤ 1.94 psig
c. Reactor Vessel Water Level - High, Level 8	≤ 53.5 inches*	≤ 55.7 inches
d. Condensate Storage Tank Level - Low	≥ 0 inches	≥ -3 inches
e. Suppression Pool Water Level - High	≤ 5.9 inches	≤ 6.5 inches
f. Manual Initiation	NA	NA

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

6. (GGNS-319) PL.

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL (a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
3. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level-Low Low, Level 2	6(c)(d)(h)	2	1, 2, 3, and #	25
b. Drywell Pressure - High	6(c)(d)(h)	2	1, 2, 3	25
c. Fuel Handling Area Ventilation Exhaust Radiation - High High	~~6(c)(h)~~ N.A.(j)	2	1, 2, 3, and *	25
d. Fuel Handling Area Pool Sweep Exhaust Radiation - High High	~~6(c)(h)~~ NA(j)	2	1, 2, 3, and *	25
e. Manual Initiation	6(f) 6(f)	1/group 1/group	1, 2, 3 *	26 25
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	8	1	1, 2, 3	27
b. Δ Flow Timer	8	1	1, 2, 3	27
c. Equipment Area Temperature - High	8	1	1, 2, 3	27
d. Equipment Area Δ Temp. - High	8	1	1, 2, 3	27
e. Reactor Vessel Water Level - Low Low, Level 2	8	2	1, 2, 3	27
f. Main Steam Line Tunnel Ambient Temperature - High	8	1	1, 2, 3	27
g. Main Steam Line Tunnel Δ Temp. - High	8	1	1, 2, 3	27
h. SLCS Initiation	8(i)	NA	1, 2, 3	27
i. Manual Initiation	8	1/group	1, 2, 3	26

6. (GGNS-319) P2.

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION
ACTION

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

ACTION 21 - Close the affected system isolation valve(s) within one hour or:

a. In OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In Operational Condition *, suspend CORE ALTERATIONS, handling of irradiated fuel in the containment and operations with a potential for draining the reactor vessel.

ACTION 22 - Restore the manual initiation function to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION 23 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 24 - Be in at least STARTUP within 6 hours.

ACTION 25 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within one hour.

ACTION 26 - Restore the manual initiation function to OPERABLE status within 8 hours or close the affected system isolation valves within the next hour and declare the affected system inoperable.

ACTION 27 - Close the affected system isolation valves within one hour and declare the affected system inoperable.

ACTION 28 - Lock the affected system isolation valves closed within one hour and declare the affected system inoperable.

NOTES

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

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

(a) See Specification 3.6.4, Table 3.6.4-1 for valves in each valve group.

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

(c) Also actuates the standby gas treatment system.

(d) Also actuates the control room emergency filtration system in the isolation mode of operation.

(e) One upscale and/or two downscale actuate the trip system.

(f) Also trips and isolates the mechanical vacuum pumps.

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

(h) Also actuates secondary containment ventilation isolation dampers and valves per Table 3.6.6.2-1.

(i) Closes only RWCU system inlet outboard valve G33-F004.

(j) Actuates the Standby Gas Treatment System and isolates Auxiliary Building penetration of the ventilation systems within the Auxiliary Building

7. (GGNS-50, 94)

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each isolation valve shown in Table 3.6.4-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.

4.6.4.2 Each automatic isolation valve shown in Table 3.6.4-1 shall be demonstrated OPERABLE during COLD SHUTDOWN or REFUELING at least once per 18 months by verifying that on an isolation test signal each automatic isolation valve actuates to its isolation position.

4.6.4.3 The isolation time of each power operated or automatic valve shown in Table 3.6.4-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

~~4.6.4.4 Each traversing in-core probe system explosive isolation valve shall be demonstrated OPERABLE:~~

~~a. At least once per 31 days by verifying the continuity of the explosive charge.~~

~~b. At least once per 18 months by removing at least one explosive squib from each explosive valve such that each explosive squib in each explosive valve will be tested at least once per 36 months, and initiating the explosive squibs. The replacement charge for the exploded squibs shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No squib shall remain in use beyond the expiration of its shelf-life and operating life, as applicable.~~

8. (GGNS-79)

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 SERVICE WATER SYSTEMS

The OPERABILITY of the service water systems ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of these systems, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits.

3/4.7.2 CONTROL ROOM EMERGENCY FILTRATION SYSTEM

The OPERABILITY of the control room emergency filtration system ensures that the control room will remain habitable for operations personnel during and following all design basis accident conditions. Cumulative operation of the system for 10 hours with the heaters OPERABLE over a 31 day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR Part 50.

~~The surveillance requirements provide adequate assurance that RCICS will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to start cooling at the earliest possible moment.~~

3/4.7.3 REACTOR CORE ISOLATION COOLING SYSTEM

The reactor core isolation cooling (RCIC) system is provided 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 requiring actuation of any of the Emergency Core Cooling System equipment. The RCIC system is conservatively required to be OPERABLE whenever reactor pressure exceeds 135 psig even though the LPCI mode of the residual heat removal (RHR) system provides adequate core cooling up to 225 psig.

The RCIC system specifications are applicable during OPERATIONAL CONDITIONS 1, 2 and 3 when reactor vessel pressure exceeds 135 psig because RCIC is the primary non-ECCS source of emergency core cooling when the reactor is pressurized.

With the RCIC system inoperable, adequate core cooling is assured by the OPERABILITY of the HPCS system and justifies the specified 14 day out-of-service period.

The surveillance requirements provide adequate assurance that RCICS will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to start cooling at the earliest possible moment.

9. (GGNS-236, 300) P1.

TABLE 3.6.4-1

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM AND VALVE NUMBER		PENETRATION NUMBER	VALVE GROUP[a]	MAXIMUM ISOLATION TIME (Seconds)
1. Automatic Isolation Valves				
a. Containment				
Main Steam Lines	B21-F028A	5(O)	1	5
Main Steam Lines	B21-F022A	5(I)	1	5
Main Steam Lines	B21-F067A-A	5(O)	1	6
Main Steam Lines	B21-F028B	6(O)	1	5
Main Steam Lines	B21-F022B	6(I)	1	5
Main Steam Lines	B21-F067B-A	6(O)	1	6
Main Steam Lines	B21-F028C	7(O)	1	5
Main Steam Lines	B21-F022C	7(I)	1	5
Main Steam Lines	B21-F067C-A	7(O)	1	6
Main Steam Lines	B21-F028D	8(O)	1	5
Main Steam Lines	B21-F022D	8(I)	1	5
Main Steam Lines	B21-F067D-A	8(O)	1	6
RHR Reactor Shutdown Cooling Suction	E12-F008-A	14(O)[c]	3	40
RHR Reactor Shutdown Cooling Suction	E12-F009-B	14(I)[c]	3	40
Steam Supply to RHR and RCIC Turbine	E51-F063-B	17(I)	4	20
Steam Supply to RHR and RCIC Turbine	E51-F064-A	17(O)	4	20
Steam Supply to RHR and RCIC Turbine	E51-F076-B	17(I)	4	20
~~RCIC and~~ RHR to Head Spray	E12-F023-B	18(O)[c]	3	90
Main Steam Line Drains	B21-F019-A	19(O)	1	15
Main Steam Line Drains	B21-F016-B	19(I)	1	15
RHR Heat Exchanger "A" to LPCI	E12-F042A-A	20(I)[c]	5	22

(a) See Specification 3.3.2, Table 3.3.2-1, for isolation signal(s) that operates each valve group.
(b) Hydrostatically tested to ASME Section XI criteria.
(c) Hydrostatically tested with water at P_a, 11.5 psig.
(d) Hydrostatically tested by pressurizing system to 1.10 P_a, 12.65 psig.
(e) Hydrostatically tested during system functional tests.
(f) Hydrostatically sealed by feedwater leakage control system. Type C test not required.

9. (GGNS-236, 300) P2.

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM AND VALVE NUMBER		PENETRATION NUMBER
b. Drywell		
Cont. Cooling Water Inlet	P42-F114-B	329(O)
Cont. Cooling Water Outlet	P42-F116-A	330(I)
Cont. Cooling Water Outlet	P42-F117-B	330(O)
Plant Serv. Water Return	P44-F076-A	331(I)
Plant Serv. Water Return	P44-F077-B	331(O)
Plant Serv. Water Supply	P44-F074-B	332(O)
Condensate Flush Connection	B33-F204	333(I)
Condensate Flush Connection	B33-F205	333(O)
3. Other Isolation Valves		
a. Containment		
Fuel Transfer Tube	F11-E015	4(I)
Cont. Leak Rate Sys.	NA	40(I)(O)
Feedwater Inlet	B21-F010A	9(I)(f)
Feedwater Inlet	B21-F032A	9(O)(f)
Feedwater Inlet	B21-F010B	10(I)(f)
Feedwater Inlet	B21-F032B	10(O)(f)
RHR "A" Suction	E12-F017A	11(O)(d)
RHR "B" Suction	E12-F017B	12(O)(d)
RHR "C" Suction	E12-F017C	13(O)(d)
RHR Shutdown Cooling Suction	E12-F308	14(I)(c)
~~RCIC &~~ RHR Head Spray	E51-F066	18(I)(c)
~~RCIC &~~ RHR Head Spray	E12 ~~E51~~-F344	18(I)(c)
RHR Heat Ex. "A" to LPCI	E12-F044A	20(I)(c)
RHR Heat Ex. "A" to LPCI	E12-F025A	20(I)(c)
RHR Heat Ex. "A" to LPCI	E12-F107A	20(I)(c)
RHR Heat Ex. "B" to LPCI	E12-F025B	21(I)(c)
RHR Heat Ex. "B" to LPCI	E12-F044B	21(I)(c)
RHR Heat Ex. "B" to LPCI	E12-F107B	21(I)(c)

9 .(GGNS-236, 300) P3.

TABLE 3.6.4-1 (Continued)

CONTAINMENT AND DRYWELL ISOLATION VALVES

SYSTEM AND VALVE NUMBER		PENETRATION NUMBER
4. Test Connections		
a. Containment		
Main Steam T/C	B21-F025A	5(0)
Main Steam T/C	B21-F025B	6(0)
Main Steam T/C	B21-F025C	7(0)
Main Steam T/C	B21-F025D	8(0)
Feedwater T/C	B21-F030A	9(0)[f]
Feedwater T/C	B21-F063A	9(0)[f]
Feedwater T/C	B21-F063B	10(0)[f]
Feedwater T/C	B21-F030B	10(0)[f]
RHR Shutdown Cool. Suction T/C	E12-F002	14(0)[c]
RCIC Steam Line T/C	E51-F072	17(0)
RHR to Head Spray T/C	E12 ~~E51~~-F342	18(0)[c]
RHR to Head Spray T/C	E12 ~~E51~~-F061	18(0)[c]
LPCI "C" T/C	E12-F056C	22(0)[c]
RHR "A" Pump Test Line T/C	E12-F322	23(0)[c]
RHR "A" Pump Test Line T/C	E12-F336	23(0)[c]
RHR "A" Pump Test Line T/C	E12-F349	23(0)[c]
RHR "A" Pump Test Line T/C	E12-F303	23(0)[c]
RHR "A" Pump Test Line T/C	E12-F310	23(0)[c]
RHR "A" Pump Test Line T/C	E12-F348	23(0)[c]
RHR"C" Pump Test Line T/C	E12-F311	24(0)[c]
RHR"C" Pump Test Line T/C	E12-F304	24(0)[c]
HPCS Discharge T/C	E22-F021	26(0)[c]
HPCS Test Line T/C	E22-F303	27(0)[c]
HPCS Test Line T/C	E22-F304	27(0)[c]
RCIC Turbine Exhaust T/C	E51-F258	24(0)[c]
RCIC Turbine Exhaust T/C	E51-F257	27(0)[c]
LPCS T/C	E21-F013	31(0)[c]
LPCS Test Line T/C	E21-F222	32(0)[c]
LPCS Test Line T/C	E21-F221	32(0)[c]

10. (GGNS-271) P1.

NO CHANGES THIS PAGE

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed the following limits:

Reactor Vessel Dome Pressure (psig)*	Maximum Insertion Times to Notch Position (Seconds) 43	29	13
950	0.31	0.81	1.44
1050	0.32	0.86	1.57

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With the maximum scram insertion time of one or more control rods exceeding the maximum scram insertion time limits of Specification 3.1.3.2 as determined by Surveillance Requirement 4.1.3.2.a or b, operation may continue provided that:

1. For all "slow" control rods, i.e., those which exceed the limits of Specification 3.1.3.2, the individual scram insertion times do not exceed the following limits:

Reactor Vessel Dome Pressure (psig)*	Maximum Insertion Times to Notch Position (Seconds) 43	29	13
950	0.38	1.09	2.09
1050	0.39	1.14	2.22

2. For "fast" control rods, i.e., those which satisfy the limits of Specification 3.1.3.2, the average scram insertion times do not exceed the following limits:

Reactor Vessel Dome Pressure (psig)*	Maximum Average Insertion Times to Notch Position (Seconds) 43	29	13
950	0.30	0.78	1.40
1050	0.31	0.84	1.53

3. The sum of "fast" control rods with individual scram insertion times in excess of the limits of ACTION a.2 and of "slow" control rods does not exceed 7.

4. No "slow" control rod, "fast" control rod with individual scram insertion time in excess of the limits of ACTION a.2, or otherwise inoperable control rod occupy adjacent locations in any direction, including the diagonal, to another such control rod.

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

*For intermediate reactor vessel dome pressure, the scram time criteria is determined by linear interpolation at each notch position.

10. (GGNS-271) P2.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

b. With a "slow" control rod(s) not satisfying ACTION ~~a.1.a~~ a.1, above:

1. Declare the "slow" control rod(s) inoperable, and
2. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with three or more "slow" control rods declared inoperable.

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

c. With the maximum scram insertion time of one or more control rods exceeding the maximum scram insertion time limits of Specification 3.1.3.2 as determined by Specification 4.1.3.2.c, operation may continue provided that:

1. "Slow" control rods, i.e., those which exceed the limits of Specification 3.1.3.2, do not make up more than 20% of the 10% sample of control rods tested.
2. Each of these "slow" control rods satisfies the limits of ACTION a.1.
3. The eight adjacent control rods surrounding each "slow" control rod are:
   a) Demonstrated through measurement within 12 hours to satisfy the maximum scram insertion time limits of Specification 3.1.3.2, and
   b) OPERABLE.
4. The total number of "slow" control rods, as determined by Specification 4.1.3.2.c, when added to the sum of ACTION a.3, as determined by Specification 4.1.3.2.a and b, does not exceed 7.

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

SURVEILLANCE REQUIREMENTS

4.1.3.2 The maximum insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig and, during single control rod scram time tests, the control rod drive pumps isolated from the accumulators:

a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS* or after a reactor shutdown that is greater than 120 days,
b. For specifically affected individual control rods following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and
c. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of POWER OPERATION.

*Except movement of SRM, IRM, or special removable detectors or normal control rod movement.

11. (GGNS-274)

TABLE 4.3.7.11-1

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE				
a. Liquid Radwaste Effluent Line	D	P	R(2)	Q(1)
~~1.~~ 2. FLOW RATE MEASUREMENT DEVICES (4)				
a. Liquid Radwaste Effluent Line	D(3)	N.A.	R	Q
b. Discharge Canal	D(3)	N.A.	R	Q

12. (GGNS-74) P1.

ELECTRICAL POWER SYSTEMS

LIMITING CONDITIONS FOR OPERATION (Continued)

ACTION:

a. For A.C. power distribution:

1. With either Division 1 or Division 2 of the above required A.C. distribution system not energized, re-energize the division within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With Division 3 of the above required A.C. distribution system not energized, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.1.

3. With one of the above required load shedding and sequencing panels inoperable, restore the inoperable panel to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. For D.C. power distribution:

1. With either Division 1 or Division 2 of the above required D.C. distribution system not energized, re-energize the division within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With Division 3 of the above required D.C. distribution system not energized, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.1.

SURVEILLANCE REQUIREMENTS

4.8.3.1.1 Each of the above required power distribution system divisions shall be determined energized at least once per 7 days by verifying correct breaker alignment ~~and voltage on the busses/MCCs/panels~~. on the busses/LCs/MCCs/panels and voltage on the busses/LCs.

4.8.3.1.2 Each of the above required load shedding and sequencing panels shall be demonstrated OPERABLE:

a. At least once per 12 hours by determining that the auto-test system is operating and is not indicating a faulted condition.

b. At least once per 31 days by performance of a manual test and verifying response within the design criteria to the following test inputs:

a) LOCA.
b) Bus undervoltage.
c) Bus undervoltage followed by LOCA.
d) LOCA followed by bus undervoltage.

12. (GGNS-74) P2.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

a. For A.C. power distribution:

1. With both Division 1 and Division 2 of the above required A.C. distribution system not energized and/or with the load shedding and sequencing panel associated with the division(s) required to be energized inoperable, suspend CORE ALTERATIONS, handling of irradiated fuel in the Auxiliary Building and Enclosure Building and operations with a potential for draining the reactor vessel.

2. With Division 3 of the above required A.C. distribution system not energized, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.2 and 3.5.3.

b. For D.C. power distribution:

1. With both Division 1 and Division 2 of the above required D.C. distribution system not energized, suspend CORE ALTERATIONS, handling of irradiated fuel in the Auxiliary Building and Enclosure Building and operations with a potential for draining the reactor vessel.

2. With Division 3 of the above required D.C. distribution system not energized, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.2 and 3.5.3.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.3.2.1 At least the above required power distribution system divisions shall be determined energized at least once per 7 days by verifying correct breaker alignment ~~and voltage on the busses/MCCs/panels~~. on the busses/LCs/MCCs/panels and voltage on the busses/LCs.

4.8.3.2.2 The above required load shedding and sequencing panel(s) shall be demonstrated OPERABLE:

a. At least once per 12 hours by determining that the auto-test system is operating and is not indicating a faulted condition.

b. At least once per 31 days by performance of a manual test and verifying response within the design criteria to the following test inputs:

a) LOCA.
b) Bus undervoltage.
c) Bus undervoltage followed by LOCA.
d) LOCA followed by bus undervoltage.

13.(GGNS-324) P1.

SURVEILLANCE REQUIREMENTS

4.1.4.2 The RPCS shall be demonstrated OPERABLE by verifying the OPERABILITY of the:

a. Rod pattern controller when THERMAL POWER is less than the low power setpoint by selecting and attempting to move an inhibited control rod:

   1. After withdrawal of the first insequence control rod for each reactor startup.

   2. As soon as the rod inhibit mode is automatically initiated at the RPCS low power setpoint. ~~< 20 of RATED THERMAL POWER.~~ 20% + 15% - 0% of Rated Thermal Power during power reduction.

   3. The first time only that a banked position, N1, N2, or N3, is reached during startup or during power reduction below the RPCS low power setpoint.

b. Rod withdrawal limiter when THERMAL POWER is greater than or equal to the low power setpoint by selecting and attempting to move a restricted control rod in excess of the allowable distance:

   1. As each power range above the RPCS low power setpoint is entered during a power increase or decrease.

   2. At least once per 31 days while operation continues within a given power range above the RPCS low power setpoint.

13. (GGNS-324) P2.

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD PATTERN CONTROL SYSTEM		
a. Low Power Setpoint	20% +15%,-0% ~~≤ 20%~~ of RATED THERMAL POWER	20% +15%,-0% ~~≤ 20%~~ of RATED THERMAL POWER
b. Intermediate Rod Withdrawal Limiter Setpoint	≤ 70% of RATED THERMAL POWER	≤ 70% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Neutron Flux-Upscale	≤ 0.66 W + 42%*	≤ 0.66 W + 45%*
b. Inoperative	NA	NA
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale Startup	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 1×10^5 cps	≤ 1.5×10^5 cps
c. Inoperative	NA	NA
d. Downscale	≥ 3 cps	≥ 2 cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 108/125 of full scale	≤ 110/125 of full scale
c. Inoperative	NA	NA
d. Downscale	≥ 5/125 of full scale	≥ 3/125 of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High	≤ 32 inches	≤ 33.5 inches
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	≤ 108% of rated flow	≤ 111% of rated flow

*The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

14.(GGNS-257)

TABLE 3.8.4.1-1

PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

DEVICE NUMBER AND LOCATION	TRIP SETPOINT (Amperes)	RESPONSE TIME (Cycles)	SYSTEM/ COMPONENT AFFECTED
a. 6.9 kV Circuit Breakers			
1103 252-1103-B	7200/45 ± ~~6400/40/±~~ 10%#	60	Reactor Recir. Pump
252-~~1003~~-C	7200/45 ± ~~6400/40/±~~ 10%#	60	Pump B33C001A
252-1205-B	7200/45 ± ~~6400/40/±~~ 10%#	60	Reactor Recir. Pump
252-1205-C	7200/45 ± ~~6400/40/±~~ 10%#	60	Pump B33C001B

b. 480 VAC Molded Case Circuit Breakers

1. Stored Energy Type SS3G3

BREAKER NUMBER	TRIP SETPOINT (Amperes)	RESPONSE TIME (Seconds)	SYSTEM/COMPONENT AFFECTED
52-12202	1200	0.05	CONTAINMENT COOLING FILTER TRAIN HEATERS (N1M41D002B-N)
52-12209	2000	0.05	CNTMT POLAR CRANE (Q1F13E001-N)
51-11502	1200	0.05	CNTMT CLG. FILTER TRAIN HEATER (N1M41D002A-N)
52-15105	2000	0.05	DRYWELL PURGE COMPRESS. (Q1E61C001A-A)
52-16204	2000	0.05	DRYWELL PURGE COMPRESS. (Q1E61C001B-B)
52-16404	1200	0.05	HYDROGEN RECOMBINER (Q1E61C003B-B)

#Primary current/setpoint.

15.(GGNS-297, 320)

TABLE 4.8.2.1-1

BATTERY SURVEILLANCE REQUIREMENTS

Parameter	CATEGORY A(1)	CATEGORY B(2)	
	Limits for each designated pilot cell	Limits for each connected cell	Allowable(3) value for each connected cell
Electrolyte Level	>Minimum level indication mark, and < ¼" above maximum level indication mark	>Minimum level indication mark, and < ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts(b)	> 2.07 volts
Specific Gravity(a)	≥ ~~1.200~~ 1.195	≥ ~~1.195~~ 1.190 Average of all connected cells > ~~1.205~~ 1.200	Not more than .020 below the average of all connected cells Average of all connected cells ≥ ~~1.195~~ 1.190

(a) Corrected for electrolyte temperature and level.
(b) May be corrected for average electrolyte temperature.
(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.
(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.
(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

16. (GGNS-302)

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level - Low Low, Level 2	≤ 13[(a)]
b. Drywell Pressure - High	≤ 13[(a)]
c. Containment and Drywell Ventilation Exhaust Radiation - High[(b)]	~~≤ 1.0*/~~ ≤ 13[(a)]**
d. Manual Initiation	NA
2. MAIN STEAM LINE ISOLATION	
a. Reactor Vessel Water Level - Low Low Low, Level 1	≤ 1.0*/≤ 13[(a)]**
b. Main Steam Line Radiation - High[(b)]	≤ 1.0*/≤ 13[(a)]**
c. Main Steam Line Pressure - Low	≤ 1.0*/≤ 13[(a)]**
d. Main Steam Line Flow - High	≤ 0.5*/≤ 13[(a)]**
e. Condenser Vacuum - Low	NA
f. Main Steam Line Tunnel Temperature - High	NA
g. Main Steam Line Tunnel Δ Temp. - High	NA
h. Manual Initiation	NA
3. SECONDARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level - Low Low, Level 2	≤ 13[(a)]
b. Drywell Pressure - High	≤ 13[(a)]
c. Fuel Handling Area Ventilation Exhaust Radiation - High High(b)	≤ 13[(a)]
d. Fuel Handling Area Pool Sweep Exhaust Radiation - High High(b)	≤ 13[(a)]
e. Manual Initiation	NA
4. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. Δ Flow - High	NA##
b. Δ Flow Timer	NA
c. Equipment Area Temperature - High	NA
d. Equipment Area Δ Temp. - High	NA
e. Reactor Vessel Water Level - Low Low, Level 2	≤ 13[(a)]
f. Main Steam Line Tunnel Ambient Temperature - High	NA
g. Main Steam Line Tunnel Δ Temp. - High	NA
h. SLCS Initiation	NA
i. Manual Initiation	NA

17.(GGNS-352,356)

TABLE 1.1

SURVEILLANCE FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
A	At least once per 366 days.
R	At least once per 18 months (550 days).
S/U	Prior to each reactor startup.
N.A.	Not applicable.
P	Completed prior to each release

18. (GGNS-362)

TABLE 3.6.6.2-1

SECONDARY CONTAINMENT VENTILATION SYSTEM AUTOMATIC ISOLATION DAMPERS/VALVES

DAMPER/VALVE FUNCTION (Number)	MAXIMUM ISOLATION TIME (Seconds)
a. Dampers	
Auxiliary Building Ventilation Supply Damper (Q1T41F006)	5
Auxiliary Building Ventilation Supply Damper (Q1T41F007)	5
Fuel Handling Area Ventilation Exhaust Damper (Q1T42F003)	5
Fuel Handling Area Ventilation Exhaust Damper (Q1T42F004)	5
Fuel Handling Area Ventilation Supply Damper ~~(Q1T42F0011)~~ (Q1T42F011)	5
Fuel Handling Area Ventilation Supply Damper ~~(Q1T42F0012)~~ (Q1T42F012)	5
Fuel Pool Sweep Ventilation Supply Damper (Q1T42F019)	5
Fuel Pool Sweep Ventilation Supply Damper (Q1T42F020)	5

19. (GGNS-102, 384) P1.

TABLE 3.3.7.12-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. RADWASTE BUILDING VENTILATION MONITORING SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm	1	*	121
b. Iodine Sampler	1	*	122
c. Particulate Sampler	1	*	122
d. Effluent System Flow Rate Measuring Device	1	*	123
e. Sampler Flow Rate Measuring Device	1	*	123
2. MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM			
a. Hydrogen Monitor	1	**	124
3. CONTAINMENT VENTILATION MONITORING SYSTEM			
a. Noble Gas Activity Monitor Providing Alarm ~~and Automatic Termination of Release~~	1	*	121 ~~125~~
b. Iodine Sampler	1	*	122
c. Particulate Sampler	1	*	122
d. Effluent System Flow Rate Monitor	1	*	123
e. Sampler Flow Rate Monitor	1	*	123

TABLE 3.3.7.12-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
6. OFFGAS PRE-TREATMENT MONITOR			
a. Noble Gas Activity Monitor	1	***	125 ~~126~~
7. OFFGAS POST-TREATMENT MONITOR			
a. Noblé Gas Activity Monitor Providing Alarm and Automatic Termination of Release	1	**	121

19.(GGNS-102, 384) P2.

TABLE 3.3.7.12-1 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

TABLE NOTATION

* At all times.

** During main condenser offgas treatment system operation.

*** During operation of the main condenser air ejector.

ACTION 121 - With the number of channels OPERABLE less than required by the Minmum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 8 hours and these samples are analyzed for gross activity within 24 hours.

ACTION 122 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided samples are continuously collected with auxiliary sampling equipment as required by Table 4.11.2.1.2-1.

ACTION 123 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent release via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 8 hours.

ACTION 124 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of main condenser offgas treatment system may continue for up to 30 days provided grab samples are collected at least once per 4 hours and analyzed within the following 4 hours.

~~ACTION 125 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, suspend release of radioactive effluents via this pathway.~~

ACTION ~~126~~ 125 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the SJAE effluent may be released to the environment for up to 72 hours provided:

a. The offgas system is not bypassed, except for filtration system bypass during plant startups, and

b. The offgas delay system noble gas activity effluent downstream monitor is OPERABLE;

Otherwise, be in at least HOT STANDBY within 12 hours.

TABLE 4.3.7.12-1

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. RADWASTE BUILDING VENTILATION MONITORING SYSTEM					
a. Noble Gas Activity Monitor - Providing Alarm	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	N.A.	*
2. MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM					
a. Hydrogen Monitor	D	N.A.	Q(4)	M	**
3. CONTAINMENT VENTILATION MONITORING SYSTEM					
a. Noble Gas Activity Monitor - Providing Alarm ~~and Automatic Termination of Release~~	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Effluent System Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	N.A.	*

19. (GGNS-102, 384) p4.

20 .(GGNS-173, 237)

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level - Low Low, Level 2	≥ -41.6 inches *	≥ -43.8 inches
b. Drywell Pressure - High	≤ 1.73 psig	≤ 1.93 psig
c. Containment and Drywell Ventilation Exhaust Radiation - High	≤ 2.0 mr/hr**	≤ 4.0 mr/hr**
d. Manual Initiation	NA	NA
2. MAIN STEAM LINE ISOLATION		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -150.3 inches*	≥ -152.5 inches
b. Main Steam Line Radiation - High	≤3.0 ~~≤ 1.5~~ x full power background	≤3.6 ~~≤ 3.0~~ x full power background
c. Main Steam Line Pressure - Low	≥ 849 psig	≥ 837 psig
d. Main Steam Line Flow - High	≤ 169 psid	≤ 176.5 psid
e. Condenser Vacuum - Low	≥ 9 inches Hg. Vacuum	≥ 8.7 inches Hg. Vacuum
f. Main Steam Line Tunnel Temperature - High	≤ 180°F**	≤ 186°F**
g. Main Steam Line Tunnel Δ Temp. - High	≤ 80°F**	≤ 83°F**
h. Manual Initiation	NA	NA
3. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level - Low Low, Level 2	≥ -41.6 inches*	≥ -43.8 inches
b. Drywell Pressure - High	≤ 1.73 psig	≤ 1.93 psig
c. Fuel Handling Area Ventilation Exhaust Radition - High High	≤ 2.0 mR/hr**	≤ 4.0 mR/hr**
d. Fuel Handling Area Pool Sweep Exhaust Radiation - High High	≤ 18 mR/hr**	≤ 35 mR/hr**
e. Manual Initiation	NA	NA

21. (GGNS-256)

TABLE 3.3.7.1-1

RADIATION MONITORING INSTRUMENTATION

INSTRUMENTATION	MINIMUM CHANNELS OPERABLE	APPLICABLE CONDITIONS	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. Component Cooling Water Radiation Monitor	1	At all times	$\leq 1 \times 10^5$ cpm/NA	~~1~~ 10 to 10^6 cpm	70
2. Standby Service Water System Radiation Monitor	1/heat exchanger train	1, 2, 3, and*	$\leq 1 \times 10^5$ cpm/NA	~~1~~ 10 to 10^6 cpm	70
3. Offgas Pre-treatment Radiation Monitor	1	1, 2	$\leq 5 \times 10^3$ mR/hr/NA	1 to 10^6 mR/hr	70
4. Offgas Post-treatment Radiation Monitor	2(a)	1, 2	$\leq 1 \times 10^5$ cpm (Hi), $\leq 1.0 \times 10^6$ cpm (Hi Hi)	~~1~~ 10 to 10^6 cpm	71
5. Carbon Bed Vault Radiation Monitor	1	1, 2	$\leq$ 2 x full power background/NA	1 to 10^6 mR/hr	72
6. Control Room Ventilation Radiation Monitor	2	1,2,3,5 and**	$\leq$4 mR/hr/ $\leq$5 mR/hr#	10^{-2} to 10^2 mR/hr	73
7. Containment and Drywell Ventilation Exhaust Radiation Monitor	3(a)	At all times	$\leq$2.0 mR/hr/ $\leq$4 mR/hr(b)#	10^{-2} to 10^2 mR/hr	74
8. Fuel Handling Area Ventilation Exhaust Radiation Monitor	3(a)	1,2,3,5 and**	$\leq$ 2mR/hr/ $\leq$4 mR/hr(d)#	10^{-2} to 10^2 mR/hr	75
9. Fuel Handling Area Pool Sweep Exhaust Radiation Monitor	3(a)	(c)	$\leq$ 18 mR/hr/ $\leq$35 mR/hr(d)#	10^{-2} to 10^2 mR/hr	75

22. (GGNS-354)

14. Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day tank of each diesel via the installed lines.

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.

16. Verifying that the following diesel generator lockout features prevent diesel generator starting and/or trip the diesel generator only when required:

    a) Generator loss of excitation.
    b) Generator reverse power.
    c) High jacket water temperature.
    d) Generator overcurrent with voltage restraint.

Engine

    e) Bus underfrequency (11 and 12 only).
    f) ~~Generator~~ bearing temperature high (11 and 12 only).
    g) Low turbo charger oil pressure (11 and 12 only).
    h) High vibration (11 and 12 only).
    i) High lube oil temperature (11 and 12 only).
    j) Low lube oil pressure (13 only).
    k) High crankcase pressure.

e. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting all three diesel generators simultaneously, during shutdown, and verifying that the three diesel generators accelerate to at least 441 rpm for diesel generators 11 and 12 and 882 rpm for diesel generator 13 in less than or equal to 13 seconds.

f. At least once per 10 years by:

    1. Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite or equivalent solution, and

    2. Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code in accordance with ASME Code Section 11, Article IWD-5000.

4.8.1.1.3 Reports - All diesel generator failures, valid or non-valid, shall be reported to the Commission pursuant to Specification 6.9.1. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests, on a per nuclear unit basis, is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

23. (GGNS-158, 292, 17) P1.

TABLE 3.8.4.2-1

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	BYPASS DEVICE (CONTINUOUS) (ACCIDENT CONDITIONS) (NO)	SYSTEM(S) AFFECTED
Q1E51F010-A	Continuous	RCIC System
Q1E51F013-A	Continuous	RCIC System
Q1E51F019-A	Continuous	RCIC System
Q1E51F022-A	Continuous	RCIC System
Q1E51F031-A	Continuous	RCIC System
Q1E51F045-A	Continuous	RCIC System
Q1E51F046-A	Continuous	RCIC System
Q1E51F059-A	Continuous	RCIC System
Q1E51F068-A	Continuous	RCIC System
Valve on Turbine Q1E51C002	~~No~~ Continuous	RCIC System
Q1B21F065A-A	No	Reactor Coolant System
Q1B21F065B-A	No	Reactor Coolant System
Q1B21F098A-~~A~~ B	No	Reactor Coolant System
Q1B21F098B-~~A~~ B	No	Reactor Coolant System
Q1B21F098C-~~A~~ B	No	Reactor Coolant System
Q1B21F098D-~~A~~ B	No	Reactor Coolant System
Q1B21F019	Continuous	Reactor Coolant System
Q1B21F067A	Continuous	Reactor Coolant System
Q1B21F067B	Continuous	Reactor Coolant System
Q1B21F067C	Continuous	Reactor Coolant System
Q1B21F067D	Continuous	Reactor Coolant System
Q1B21F016	Continuous	Reactor Coolant System
Q1B21F147A	Continuous	MSL Drain Post LOCA Leakage Control
Q1B21F147B	Continuous	MSL Drain Post LOCA Leakage Control
Q1B33F019	Continuous	Recirculation System
Q1B33F020	Continuous	Recirculation System
Q1B33F125	Continuous	Recirculation System
Q1B33F126	Continuous	Recirculation System
Q1B33F127	Continuous	Recirculation System
Q1B33F128	Continuous	Recirculation System
Q1D23F591B	*	Drywell Monitoring System
Q1D23F592A	*	Drywell Monitoring System
Q1D23F593B	*	Drywell Monitoring System
Q1D23F594A	*	Drywell Monitoring System
Q1E12F040	Continuous	RHR System
Q1E12F023	Continuous	RHR System
Q1E12F006A	Continuous	RHR System
Q1E12F052A	Continuous	RHR System
Q1E12F008	Continuous	RHR System
Q1E12F074A	Continuous	RHR System
Q1E12F026A	Continuous	RHR System
Q1E12F082A	No	RHR System
Q1E12F082B	No	RHR System

23.(GGNS-158,292,17) P2.

TABLE 3.8.4.2-1 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	BYPASS DEVICE (CONTINUOUS) (ACCIDENT CONDITIONS) (NO)	SYSTEM(S) AFFECTED
Q1E12F290A	Continuous	RHR System
Q1E12F047A	Continuous	RHR System
Q1E12F027A	Continuous	RHR System
Q1E12F073A	Continuous	RHR System
Q1E12F346	Continuous	RHR System
Q1E12F024A	Continuous	RHR System
Q1E12F087A	Continuous	RHR System
Q1E12F048A	Continuous	RHR System
Q1E12F042A	Continuous	RHR System
Q1E12F004A	Continuous	RHR System
Q1E12F003A	Continuous	RHR System
Q1E12F011A	Continuous	RHR System
Q1E12F053A	Continuous	RHR System
Q1E12F037A	Continuous	RHR System
Q1E12F028A	Continuous	RHR System
Q1E12F064A	Continuous	RHR System
Q1E12F290B	Continuous	RHR System
Q1E12F004C	Continuous	RHR System
Q1E12F021	Continuous	RHR System
Q1E12F064C	Continuous	RHR System
Q1E12F042C	Continuous	RHR System
Q1E12F048B	Continuous	RHR System
Q1E12F049	Continuous	RHR System
Q1E12F037B	Continuous	RHR System
Q1E12F053B	Continuous	RHR System
Q1E12F074B	Continuous	RHR System
Q1E12F042B	Continuous	RHR System
Q1E12F064B	Continuous	RHR System
Q1E12F096	Continuous	RHR System
Q1E12F094	Continuous	RHR System
Q1E12F006B	Continuous	RHR System
Q1E12F011B	Continuous	RHR System
Q1E12F052B	Continuous	RHR System
Q1E12F047B	Continuous	RHR System
Q1E12F027B	Continuous	RHR System
Q1E12F004B	Continuous	RHR System
Q1E12F087B	Continuous	RHR System
Q1E12F003B	Continuous	RHR System
Q1E12F026B	Continuous	RHR System
Q1E12F024B	Continuous	RHR System
Q1E12F028B	Continuous	RHR System
Q1E12F009	Continuous	RHR System
Q1E12F073B	Continuous	RHR System
Q1C11F083	No	CRD Hydraulic System
Q1C11F322	Continuous	CRD Hydraulic System
Q1C41F001A	Continuous	Standby Liquid Control
Q1C41F001B	Continuous	Standby Liquid Control

23. (GGNS-158, 292, 17) P3.

TABLE 3.8.4.2-1 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	BYPASS DEVICE (CONTINUOUS) (ACCIDENT CONDITIONS) (NO)	SYSTEM(S) AFFECTED
Q1E21F001	Continuous	LPCS System
Q1E21F011	Continuous	LPCS System
Q1E21F012	Continuous	LPCS System
Q1E21F005	Continuous	LPCS System
Q1E30F002A	Continuous	Suppression Pool Makeup System
Q1E30F591A	*	Suppression Pool Makeup System
Q1E30F592A	*	Suppression Pool Makeup System
Q1E30F593A	*	Suppression Pool Makeup System
Q1E30F594A	*	Suppression Pool Makeup System
Q1E30F001A	Continuous	Suppression Pool Makeup System
Q1E30F001B	Continuous	Suppression Pool Makeup System
Q1E30F002B	Continuous	Suppression Pool Makeup System
Q1E30F591B	*	Suppression Pool Makeup System
Q1E30F592B	*	Suppression Pool Makeup System
Q1E30F593B	*	Suppression Pool Makeup System
Q1E30F594B	*	Suppression Pool Makeup System
Q1E31F100A	Continuous	Fuel Pool Cooling & Cleanup System
Q1E31F100B	Continuous	Fuel Pool Cooling & Cleanup System
Q1E32F001A	Continuous	MSIV - LCS
Q1E32F001E	Continuous	MSIV - LCS
Q1E32F003A	Continuous	MSIV - LCS
Q1E32F003E	Continuous	MSIV - LCS
Q1E32F003J	Continuous	MSIV - LCS
Q1E32F003N	Continuous	MSIV - LCS
Q1E32F001J	Continuous	MSIV - LCS
Q1E32F001N	Continuous	MSIV - LCS
Q1E32F002A	Continuous	MSIV - LCS
Q1E32F002E	Continuous	MSIV - LCS
Q1E32F002J	Continuous	MSIV - LCS
Q1E32F002N	Continuous	MSIV - LCS
Q1E32F006	Continuous	MSIV - LCS
Q1E32F007	Continuous	MSIV - LCS
Q1E32F008	Continuous	MSIV - LCS
Q1E32F009	Continuous	MSIV - LCS
Q1E38F001A	Continuous	Feedwater LCS
Q1E38F001B	Continuous	Feedwater LCS
Q1E51F064	Continuous	RCIC System
Q1E51F063	Continuous	RCIC System
Q1E51F076	Continuous	RCIC System
Q1E51F077	Continuous	RCIC System
Q1E51F078	Continuous	RCIC System
Q1E22F001	Continuous	HPCS System
Q1E22F004	Continuous	HPCS System
Q1E22F010	Continuous	HPCS System
Q1E22F011	Continuous	HPCS System
Q1E22F012	Continuous	HPCS System
Q1E22F015	Continuous	HPCS System
Q1E22F023	Continuous	HPCS System

23. (GGNS - 158, 292, 17) P4.

TABLE 3.8.4.2-1 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	BYPASS DEVICE (CONTINUOUS) (ACCIDENT CONDITIONS) (NO)	SYSTEM(S) AFFECTED
Q1E61F595A	*	Combustible Gas Control System
Q1E61F596A	*	Combustible Gas Control System
Q1E61F597A	*	Combustible Gas Control System
Q1E61F598A	*	Combustible Gas Control System
Q1E61F595C	*	Combustible Gas Control System
Q1E61F596C	*	Combustible Gas Control System
Q1E61F597C	*	Combustible Gas Control System
Q1E61F598C	*	Combustible Gas Control System
Q1E61F595B	*	Combustible Gas Control System
Q1E61F596B	*	Combustible Gas Control System
Q1E61F597B	*	Combustible Gas Control System
Q1E61F598B	*	Combustible Gas Control System
Q1E61F595D	*	Combustible Gas Control System
Q1E61F596D	*	Combustible Gas Control System
Q1E61F597D	*	Combustible Gas Control System
Q1E61F598D	*	Combustible Gas Control System
Q1E61F003A	Continuous	Combustible Gas Control System
Q1E61F005A	Continuous	Combustible Gas Control System
Q1E61F003B	Continuous	Combustible Gas Control System
Q1E61F005B	Continuous	Combustible Gas Control System
Q1G33F251	Continuous	RWCU System
Q1G33F253	Continuous	RWCU System
Q1G33F004	Continuous	RWCU System
Q1G33F039	Continuous	RWCU System
Q1G33F034	Continuous	RWCU System
Q1G33F054	Continuous	RWCU System
Q1G33F028	Continuous	RWCU System
Q1G33F053	Continuous	RWCU System
Q1G33F040	Continuous	RWCU System
Q1G33F001	Continuous	RWCU System
Q1G33F250	Continuous	RWCU System
Q1G33F252	Continuous	RWCU System
Q1G41F028	Continuous	Spent Fuel Pool Cooling and Cleanup System
Q1G41F029	Continuous	Spent Fuel Pool Cooling and Cleanup System
Q1G41F044	Continuous	Spent Fuel Pool Cooling and Cleanup System
Q1G41F043	No	Spent Fuel Pool Cooling and Cleanup System
Q1G41F021	No	Spent Fuel Pool Cooling and Cleanup System
Q1M71F595	*	Containment/Drywell I&C
Q1M71F591A	*	Containment/Drywell I&C
Q1M71F593A	*	Containment/Drywell I&C
Q1M71F592B	*	Containment/Drywell I&C
Q1M71F591B	*	Containment/Drywell I&C
Q1M71F592A	*	Containment/Drywell I&C
Q1M71F594	*	Containment/Drywell I&C
Q1P21F017	Continuous	Makeup Water Treatment System
Q1P21F018	Continuous	Makeup Water Treatment System

23. (GGNS-158, 292, 17) P5.

TABLE 3.8.4.2-1 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	BYPASS DEVICE (CONTINUOUS) (ACCIDENT CONDITIONS) (NO)	SYSTEM(S) AFFECTED
Q1P41F237	Continuous	SSW System
Q1P41F018	Continuous	SSW System
Q1P41F241	Continuous	SSW System
Q1P41F238	Continuous	SSW System
QSP41F081A ~~Q1P41F081A~~	Continuous	SSW System
QSP41F064A ~~Q1P41F064A~~	Continuous	SSW System
Q1P41F068A	Continuous	SSW System
Q1P41F014A	Continuous	SSW System
Q1P41F159A	Continuous	SSW System
Q1P41F160A	Continuous	SSW System
Q1P41F113	Continuous	SSW System
Q1P41F168A	Continuous	SSW System
Q1P41F001A	Continuous	SSW System
Q1P41F016A	Continuous	SSW System
Q1P41F015A	Continuous	SSW System
Q1P41F006A	Continuous	SSW System
Q1P41F005A	Continuous	SSW System
Q1P41F007A	Continuous	SSW System
QSP41F074A ~~Q1P41F074A~~	Continuous	SSW System
QSP41F066A ~~Q1P41F066A~~	Continuous	SSW System
QSP41F125 ~~Q1P41F125~~	Continuous	SSW System
Q1P41F018B	Continuous	SSW System
Q1P41F160B	Continuous	SSW System
Q1P41F159B	Continuous	SSW System
Q1P41F168B	Continuous	SSW System
QSP41F154 ~~Q1P41F154~~	Accident Conditions	SSW System
QSP41F155A ~~Q1P41F155A~~	Accident Conditions	SSW System
Q1P41F068B	Continuous	SSW System
QSP41F155B ~~Q1P41F155B~~	Accident Conditions	SSW System
Q1P41F014B	Continuous	SSW System
QSP41F064B ~~Q1P41F064B~~	Continuous	SSW System
QSP41F081B ~~Q1P41F081B~~	Continuous	SSW System
Q1P41F006B	Continuous	SSW System
Q1P41F007B	Continuous	SSW System
Q1P41F001B	Continuous	SSW System
Q1P41F016B	Continuous	SSW System
Q1P41F005B	Continuous	SSW System
Q1P41F015B	Continuous	SSW System
QSP41F066B ~~Q1P41F066B~~	Continuous	SSW System
QSP41F074B ~~Q1P41F074B~~	Continuous	SSW System
QSP41F189 ~~Q1P41F189~~	Continuous	SSW System
Q1P41F011	Continuous	SSW System
Q1P41F119A	No	SSW System
Q1P41F119B	No	SSW System
Q1P41F121A	No	SSW System
Q1P41F121B	No	SSW System
Q1P41F122A	No	SSW System
Q1P41F122B	No	SSW System

23 .(GGNS-158, 292, 17) P6. TABLE 3.8.4.2-1 (Continued)

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	BYPASS DEVICE (CONTINUOUS) (ACCIDENT CONDITIONS) (NO)	SYSTEM(S) AFFECTED
QSZ51F007	CONTINUOUS	CONTROL ROOM HVAC
QSZ51F008	CONTINUOUS	CONTROL ROOM HVAC
QSZ51F014	CONTINUOUS	CONTROL ROOM HVAC
QSZ51F016	CONTINUOUS	CONTROL ROOM HVAC
Q1P42F067	Continuous	CCW System
Q1P42F116	Continuous	CCW System
Q1P42F028A	Continuous	CCW System
Q1P42F032A	Continuous	CCW System
Q1P42F201A	Continuous	CCW System
Q1P42F204	Continuous	CCW System
Q1P42F205	Continuous	CCW System
Q1P42F105	Continuous	CCW System
Q1P42F200A	Continuous	CCW System
Q1P42F203	Continuous	CCW System
Q1P42F117	Continuous	CCW System
Q1P42F114	Continuous	CCW System
Q1P42F068	Continuous	CCW System
Q1P42F200B	Continuous	CCW System
Q1P42F028B	Continuous	CCW System
Q1P42F201B	Continuous	CCW System
Q1P42F032B	Continuous	CCW System
Q1P42F066	Continuous	CCW System
Q1P44F053	Continuous	Plant SW System
Q1P44F069	Continuous	Plant SW System
Q1P44F076	Continuous	Plant SW System
Q1P44F070	Continuous	Plant SW System
Q1P44F074	Continuous	Plant SW System
Q1P44F077	Continuous	Plant SW System
Q1P44F042	Continuous	Plant SW System
Q1P44F054	Continuous	Plant SW System
Q1P44F067	Continuous	Plant SW System
Q1P45F096	Continuous	Floor & Eqpt. Drain System
Q1P45F097	Continuous	Floor & Eqpt. 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

*Manual bypass of thermal overload protection of manually controlled valve.

24.(GGNS-501)

TABLE 3.3.7.5-1 (Continued)
ACCIDENT MONITORING INSTRUMENTATION

ACTION STATEMENTS

ACTION 80 -

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Required Number of Channels shown in Table 3.3.7.5-1, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3.7.5-1, or be in at least HOT SHUTDOWN within the next 12 hours. restore the inoperable channel(s) to OPERABLE status within 48 hours...

ACTION 81 -

With the number of OPERABLE accident monitoring instrumentation channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

a. Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

b. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

25. (GGNS-452)

REFUELING OPERATIONS

3/4.9.12 HORIZONTAL FUEL TRANSFER SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 The horizontal fuel transfer system (HFTS) may be in operation provided that:

a. The room through which the transfer system penetrates is sealed.

b. All interlocks with the refueling and fuel handling platforms are OPERABLE.

c. All HFTS primary carriage position indicators are OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 4* and 5*.

ACTION:

With the requirements of the above specification not satisfied, suspend HFTS operation with the HFTS at either the Spent Fuel Building pool or the Reactor Containment Building pool terminal point.

SURVEILLANCE REQUIREMENTS

4.9.12 Within ~~4~~ 24 hours prior to the operation of HFTS and at least once per ~~12 hours~~ 7 days thereafter, verify that:

a. All interlocks with the refueling and fuel handling platforms are OPERABLE.

b. All HFTS primary carriage position indicators are OPERABLE.

*When the reactor mode switch is in the Refuel position.

26. (GGNS-416, 417, 433)

3/4.9 REFUELING OPERATIONS

3/4.9.1 REACTOR MODE SWITCH

LIMITING CONDITION FOR OPERATION

3.9.1 The reactor mode switch shall be OPERABLE and locked in the Shutdown or Refuel position. When the reactor mode switch is locked in the Refuel position:

a. A control rod shall not be withdrawn unless the Refuel position one-rod-out interlock is OPERABLE.

b. CORE ALTERATIONS shall not be performed using equipment associated with a Refuel position interlock unless at least the following associated Refuel position interlocks are OPERABLE for such equipment.

1. All rods in.
2. Refuel platform position.
3. Refuel platform main ~~hoists~~ fuel-loaded.
4. ~~Fuel grapple position.~~
5. ~~Source range monitor countrate.~~

APPLICABILITY: OPERATIONAL CONDITION 5* #.

ACTION:

a. With the reactor mode switch not locked in the Shutdown or Refuel position as specified, suspend CORE ALTERATIONS and lock the reactor mode switch in the Shutdown or Refuel position.

b. With the one-rod-out interlock inoperable, lock the reactor mode switch in the Shutdown position.

c. With any of the above required Refuel position equipment interlocks inoperable, suspend CORE ALTERATIONS with equipment associated with the inoperable Refuel position equipment interlock.

* See Special Test Exceptions 3.10.1 and 3.10.3.##

The reactor shall be maintained in OPERATIONAL CONDITION 5 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

The reactor mode switch may be placed in the Run or Startup/Hot Standby position to test the switch interlock functions provided that all control rods are verified to remain fully inserted by a second licensed operator or other technically qualified member of the unit technical staff.

27. (GGNS-303) P1.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	VALVE GROUPS OPERATED BY SIGNAL (a)	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (b)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level-Low Low, Level 2	6, 7, 8, 10(c)(d)	2	1, 2, 3 and #	20
b. Drywell Pressure - High	5, 6, 7, 9(c)(d)	2	1, 2, 3	20
c. Containment and Drywell Ventilation Exhaust Radiation - High High	7	2(e)	1, 2, 3 and *	21
d. Manual Initiation	5, 6, 7, 8, 9, 10	2/group	1, 2, 3 and *#	22
2. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level-Low Low Low, Level 1	1, 5	2	1, 2, 3	20
b. Main Steam Line Radiation - High	1, 10(f)	1/line	1, 2, 3	23
c. Main Steam Line Pressure - Low	1	1/line	1	24
d. Main Steam Line Flow - High	1	2/line(g)	1, 2, 3	23
e. Condenser Vacuum - Low	1	2	1, 2, 3	23
f. Main Steam Line Tunnel Temperature - High	1	2	1, 2, 3	23
g. Main Steam Line Tunnel Δ Temp.- High	1	2	1, 2, 3	23
h. Manual Initiation	1, 5, 10	2/group	1, 2, 3	22

27. (GGNS-303) P 2

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level - Low Low, Level 2	≥ -41.6 inches *	≥ -43.8 inches
b. Drywell Pressure - High	≤ 1.73 psig	≤ 1.93 psig
c. Containment and Drywell Ventilation Exhaust Radiation - High High	≤ 2.0 mr/hr**	≤ 4.0 mr/hr**
d. Manual Initiation	NA	NA
2. MAIN STEAM LINE ISOLATION		
a. Reactor Vessel Water Level - Low Low Low, Level 1	≥ -150.3 inches*	≥ -152.5 inches
b. Main Steam Line Radiation - High	≤ 1.5 x full power background	≤ 3.0 x full power background
c. Main Steam Line Pressure - Low	≥ 849 psig	≥ 837 psig
d. Main Steam Line Flow - High	≤ 169 psid	≤ 176.5 psid
e. Condenser Vacuum - Low	≥ 9 inches Hg. Vacuum	≥ 8.7 inches Hg. Vacuum
f. Main Steam Line Tunnel Temperature - High	≤ 180°F**	≤ 186°F**
g. Main Steam Line Tunnel Δ Temp. - High	≤ 80°F**	≤ 83°F**
h. Manual Initiation	NA	NA
3. SECONDARY CONTAINMENT ISOLATION		
a. Reactor Vessel Water Level - Low Low, Level 2	≥ -41.6 inches*	≥ -43.8 inches
b. Drywell Pressure - High	≤ 1.73 psig	≤ 1.93 psig
c. Fuel Handling Area Ventilation Exhaust Radition - High High	≤ 2.0 mR/hr**	≤ 4.0 mR/hr**
d. Fuel Handling Area Pool Sweep Exhaust Radiation - High High	≤ 18 mR/hr**	≤ 35 mR/hr**
e. Manual Initiation	NA	NA

27.(GG NS-303) P3

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level - Low Low, Level 2	$\leq 13^{(a)}$
b. Drywell Pressure - High	$\leq 13^{(a)}$
c. Containment and Drywell Ventilation Exhaust Radiation - High ~~(b)~~ High(b)	$\leq 1.0^{*}/\leq 13^{(a)**}$
d. Manual Initiation	NA
2. MAIN STEAM LINE ISOLATION	
a. Reactor Vessel Water Level - Low Low Low, Level 1	$\leq 1.0^{*}/\leq 13^{(a)**}$
b. Main Steam Line Radiation - High(b)	$\leq 1.0^{*}/\leq 13^{(a)**}$
c. Main Steam Line Pressure - Low	$\leq 1.0^{*}/\leq 13^{(a)**}$
d. Main Steam Line Flow - High	$\leq 0.5^{*}/\leq 13^{(a)**}$
e. Condenser Vacuum - Low	NA
f. Main Steam Line Tunnel Temperature - High	NA
g. Main Steam Line Tunnel Δ Temp. - High	NA
h. Manual Initiation	NA
3. SECONDARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level - Low Low, Level 2	$\leq 13^{(a)}$
b. Drywell Pressure - High	$\leq 13^{(a)}$
c. Fuel Handling Area Ventilation Exhaust Radiation - High High(b)	$\leq 13^{(a)}$
d. Fuel Handling Area Pool Sweep Exhaust Radiation - High High(b)	$\leq 13^{(a)}$
e. Manual Initiation	NA
4. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. Δ Flow - High	NA##
b. Δ Flow Timer	NA
c. Equipment Area Temperature - High	NA
d. Equipment Area Δ Temp. - High	NA
e. Reactor Vessel Water Level - Low Low, Level 2	$\leq 13^{(a)}$
f. Main Steam Line Tunnel Ambient Temperature - High	NA
g. Main Steam Line Tunnel Δ Temp. - High	NA
h. SLCS Initiation	NA
i. Manual Initiation	NA

27. (GGNS-303) P4.

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low Low, Level 2	S	M	R	1, 2, 3 and #
b. Drywell Pressure - High	S	M	R	1, 2, 3
c. Containment and Drywell Ventilation Exhaust Radiation - High High	S	M	R	1, 2, 3 and *
d. Manual Initiation	NA	M(a)	NA	1, 2, 3 and *#
2. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level - Low Low Low, Level 1	S	M	R	1, 2, 3
b. Main Steam Line Radiation - High	S	M	R	1, 2, 3
c. Main Steam Line Pressure - Low	S	M	R	1
d. Main Steam Line Flow - High	S	M	R	1, 2, 3
e. Condenser Vacuum - Low	S	M	R	1, 2**, 3**
f. Main Steam Line Tunnel Temperature - High	S	M	R	1, 2, 3
g. Main Steam Line Tunnel Δ Temp. - High	S	M	R	1, 2, 3
h. Manual Initiation	NA	M(a)	NA	1, 2, 3

28. (GGNS-249)

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.

* Applicable When the system is required to be OPERABLE~~, after being manually realigned, as applicable,~~ 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) Manual initiation test shall include verification of the OPERABILITY of the LPCS and LPCI injection valve interlocks.

(d) This calibration shall consist of the CHANNEL CALIBRATION of the LPCS and LPCI injection valve interlocks with the interlock setpoint verified to be $\leq$ 150 psig.

29.(GGNS-99)

TABLE 3.3.2-1 (Continued)
ISOLATION ACTUATION INSTRUMENTATION
ACTION

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

ACTION 21 - Close the affected system isolation valve(s) within one hour or:

a. In OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In Operational Condition *, suspend CORE ALTERATIONS, handling of irradiated fuel in the containment and operations with a potential for draining the reactor vessel.

ACTION 22 - Restore the manual initiation function to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION 23 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 24 - Be in at least STARTUP within 6 hours.

ACTION 25 - Establish SECONDARY CONTAINMENT INTEGRITY with the standby gas treatment system operating within one hour.

ACTION 26 - Restore the manual initiation function to OPERABLE status within 8 hours or close the affected system isolation valves within the next hour and declare the affected system inoperable.

ACTION 27 - Close the affected system isolation valves within one hour and declare the affected system inoperable.

ACTION 28 - Lock the affected system isolation valves closed within one hour and declare the affected system inoperable.

NOTES

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

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

(a) See Specification 3.6.4, Table 3.6.4-1 for valves in each valve group.

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

(c) Also actuates the standby gas treatment system.

(d) Also actuates the control room emergency filtration system in the isolation mode of operation.

(e) One upscale and/or two downscale actuate the trip system.

(f) Also trips and isolates the mechanical vacuum pumps.

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

(h) Also actuates secondary containment ventilation isolation dampers and valves per Table 3.6.6.2-1.

(i) Closes only RWCU system ~~inlet outboard~~ isolation valves ~~G33-F004~~. G33-F001, G33-F004, and G33-F251.

30. (GGNS-470) P1.

TABLE 3.3.7.3-1

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM INSTRUMENTS OPERABLE
a. Wind Speed	
1. Elev. 33 ft and 162 ft	1 each
b. Wind Direction	
1. Elev. 33 ft and 162 ft	1 each
c. Air Temperature	
1. Elev. 33 ft ~~and 162 ft~~	1 each
d. Air Temperature Difference	
1. Elev. 33/162 ft	1

30. (GGNS-470) P2.

TABLE 4.3.7.3-1

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
a. Wind Speed		
1. Elev. 33 ft and 162 ft	D	SA
b. Wind Direction		
1. Elev. 33 ft and 162 ft	D	SA
c. Air Temperature		
1. Elev. 33 ft ~~and 162 ft~~	D	SA
d. Air Temperature Difference		
1. Elev. 33/162 ft	D	SA

31. (GGNS-457) p1.

TABLE 4.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. ROD PATTERN CONTROL SYSTEM				
a. Low Power Setpoint	NA	S/U(b)(e), D(c)(e), M(d)(e)	Q	1, 2
b. Intermediate Rod Withdrawal Limiter Setpoint	NA	S/U(b)(e), D(c)(e), M(d)(e)	Q	1, 2
2. APRM				
a. Flow Biased Neutron Flux-Upscale	NA	S/U(b),M	W(f)(g), SA	1
b. Inoperative	NA	S/U(b),M	NA	1, 2, 5
c. Downscale	NA	S/U(b),M	W(h), SA	1
d. Neutron Flux - Upscale, Startup	NA	S/U(b),M	Q	2, 5
3. SOURCE RANGE MONITORS				
a. Detector not full in	NA	S/U(b),W	NA	2, 5
b. Upscale	NA	S/U(b),W	Q	2, 5
c. Inoperative	NA	S/U(b),W	NA	2, 5
d. Downscale	NA	S/U(b),W	Q	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	S/U(b),W	NA	2, 5
b. Upscale	NA	S/U(b),W	Q	2, 5
c. Inoperative	NA	S/U(b),W	NA	2, 5
d. Downscale	NA	S/U(b),W	Q	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	NA	M	R	1, 2, 5*
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	NA	S/U(b),M	Q	1

TABLE 4.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES:

a. Neutron detectors may be excluded from CHANNEL CALIBRATION.

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

c. Within one hour prior to control rod movement, unless performed within the previous 24 hours, and as each power range above the RPCS low power setpoint is entered for the first time during any 24 hour period during power increase or decrease.

d. At least once per 31 days while operation continues within a given power range above the RPCS low power setpoint.

e. Includes reactor manual control multiplexing system input.

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

f. 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 is ≥ 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.

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

h. This calibration shall consist of verifying the trip set point only.

32.(GGNS-427) P1.

TABLE 3.3.7.2-1

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENTS OPERABLE
1. Triaxial Strong Motion Accelerometer		
a. Containment foundation	0.001 to 1.0g	1
b. Drywell	0.001 to 1.0g	1
c. SGTS Filter Train	0.001 to 1.0g	1
d. SSW Pump House A	0.001 to 1.0g	1
e. Free Field	0.001 to 1.0g	1
2. Triaxial Peak Recording Accelerograph		
a. Containment Dome	0.01 to 2g	1
b. Auxiliary Building Foundation	0.01 to 2g	1
c. Diesel Generator 11	0.01 to 2g	1
d. Control Building Foundation	0.01 to 2g	1
e. Control Room	0.01 to 2g	1
f. Reactor Vessel Support	0.01 to 2g	1
g. Reactor Recirc. Piping	0.01 to 2g	1
h. Main Steam Piping	0.01 to 2g	1
i. LPCS Spray Line	0.01 to 2g	1
j. HPCS Spray Line	0.01 to 2g	1
k. SSW Pump House B	0.01 to 2g	1
3. Triaxial Seismic Switches		
a. Containment Foundation (SSE)	0.025 to 0.25g	1*
b. Containment Foundation (OBE)	0.025 to 0.25g	1*
c. Drywell (SSE)	0.025 to 0.25g	1*
d. Drywell (OBE)	0.025 to 0.25g	1*
4. Vertical Seismic Trigger ~~Recorders~~		
a. Containment Foundation	0.005 to 0.05g	1*
5. Horizontal Seismic Trigger		
a. Drywell	0.005 to 0.05g	1*

*With control room annunciation.

32. (GGNS-427) P2.

TABLE 4.3.7.2-1

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Triaxial Strong Motion Accelerometer			
a. Containment Foundation	M	SA	R
b. Drywell	M	SA	R
c. SGTS Filter Train	M	SA	R
d. SSW Pump House A	M	SA	R
e. Free Field	M	SA	R
2. Triaxial Peak Recording Accelerograph			
a. Containment Dome	NA	NA	R
b. Auxiliary Building Foundation	NA	NA	R
c. Diesel Generator 11	NA	NA	R
d. Control Building Foundation	NA	NA	R
e. Control Room	NA	NA	R
f. Reactor Vessel Support	NA	NA	R
g. Reactor Recirc. Piping	NA	NA	R
h. Main Steam Piping	NA	NA	R
i. LPCS Spray Line	NA	NA	R
j. HPCS Spray Line	NA	NA	R
k. SSW Pump House B	NA	NA	R
3. Triaxial Seismic Switches			
a. Containment Foundation (SSE)	M	SA	R
b. Containment Foundation (OBE)	M	SA	R
c. Drywell (SSE)	M	SA	R
d. Drywell (OBE)	M	SA	R
4. Vertical Seismic Trigger ~~Recorders~~			
a. Containment Foundation	M	SA	R
5. Horizontal Seismic Trigger			
a. Drywell	M	SA	R

33. B

6.5.2 SAFETY REVIEW COMMITTEE (SRC)

FUNCTION

6.5.2.1 The SRC shall function to provide independent review and audit of designated activities in the areas of:

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

h. quality assurance practices

COMPOSITION

6.5.2.2 The SRC shall be composed of the:

Chairman:	Assistant Vice President for Nuclear Production
Member:	Manager of Nuclear Plant Engineering
Member:	Manager of Quality Assurance
Member:	Manager of System Nuclear Operations, Middle South Services, Inc.
Member:	Nuclear Plant Manager
Member:	Manager of Nuclear Services
Member:	Corporate Health Physicist
Member:	Principal Engineer, Operations Analysis
Member ~~*~~	Advisor to the Assistant Vice-President, Nuclear Operations

Two additional voting members shall be consultants to Mississippi Power and Light Company consistent with the recommendations of the Advisory Committee on Reactor Safeguards letter, Mark to Palladino dated October 20, 1981.

The SRC members shall hold a Bachelor's degree in an engineering or physical science field or equivalent experience and a minimum of five years of technical experience of which a minimum of three years shall be in one or more of the disciplines of 6.5.2.1a through h. In the aggregate, the membership of the committee shall provide specific practical experience in the majority of the disciplines of 6.5.2.1a through h.

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the SRC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in SRC activities at any one time.

~~*Non voting member.~~

34. B

RADIOACTIVE EFFLUENTS

LIQUID WASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.1.3 The liquid radwaste system components as specified in the ODCM shall be OPERABLE. The appropriate portions of the system shall be used to reduce the radioactive materials in liquid wastes prior to their discharge when the cumulative projected dose due to the liquid effluent from the site (see Figure ~~5.1.4-1~~) in a 31 day period would exceed 0.06 mrem to the total body or 0.2 mrem to any organ.

5.1.3-1 APPLICABILITY: At all times.

ACTION:

a. With the liquid radwaste treatment system inoperable for more than 31 days or with radioactive liquid waste being discharged without treatment and in excess of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report which includes the following information:

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

b. The provisions of Specifications 3.0.3, 3.0.4 and 6.9.1.11 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.3.1 Doses due to liquid releases to unrestricted areas shall be projected at least once per 31 days, in accordance with the ODCM.

4.11.1.3.2 The liquid radwaste system components specified in the ODCM shall be demonstrated OPERABLE by operating the liquid radwaste treatment system equipment for at least 30 minutes at least once per 92 days unless the liquid radwaste system has been utilized to process radioactive liquids during the previous 92 days.

35.B

AUTHORITY

6.5.2.9 The SRC shall report to and advise the Senior Vice President - Nuclear on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.2.8.

RECORDS

6.5.2.10 Records of SRC activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each SRC meeting shall be prepared, approved and forwarded to the Senior Vice President - Nuclear within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.7 AND 6.5.2.8 above, shall be ~~prepared, approved and forwarded to the Senior Vice President - Nuclear within 14 days following completion of the review.~~ DOCUMENTED IN THE MINUTES OF SRC MEETINGS AND

~~c. Audit reports encompassed by Section 6.5.2.8 above, shall be~~ forwarded to the Senior Vice President - Nuclear FOR REVIEW. ~~and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.~~

6.5.3 TECHNICAL REVIEW AND CONTROL

ACTIVITIES

6.5.3.1 Activities which affect nuclear safety shall be conducted as follows:

a. Procedures required by Technical Specification 6.8 and other procedures which affect plant nuclear safety, and changes thereto, shall be prepared, reviewed and approved. Each such procedure or procedure change shall be reviewed by an individual/group other than the individual/group which prepared the procedure or procedure change, but who may be from the same organization as the individual/group which prepared the procedure or procedure change. Procedures other than Administrative Procedures shall be approved as delineated in writing by the Plant Manager. The Plant Manager shall approve administrative procedures, security implementing procedures and emergency plant implementing procedures. Temporary approval to procedures which clearly do not change the intent of the approved procedures may be made by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License. For changes to procedures which may involve a change in intent of the approved procedures, the person authorized above to approve the procedure shall approve the change.

b. Proposed changes or modifications to plant nuclear safety-related structures, systems and components shall be reviewed as designated by the Plant Manager. Each such modification shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modifications. Implementation of proposed modifications to plant nuclear safety-related structures, systems and components shall be approved by the Plant Manager.