MISSISSIPPI POWER & LIGHT COMPANY
Helping Build Mississippi
P. O. BOX 1640, JACKSON, MISSISSIPPI 39205

August 1, 1983

NUCLEAR PRODUCTION DEPARTMENT

U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D.C. 20555

Attention: Mr. Harold R. Denton, Director

Dear Mr. Denton:

SUBJECT: Grand Gulf Nuclear Station
Unit 1
Docket No. 50-416
License No. NPF-13
File 0260/L-860.0
Transmittal of Proposed
Changes to Grand Gulf
Technical Specifications
AECM-83/0422

Consistent with NRC direction provided to Mississippi Power & Light (MP&L) in the "Confirmation of Action" letter, Mr. J. P. O'Reilly to Mr. D. C. Lutken, dated October 20, 1982 (MAEC-82/242), MP&L is providing for your review and approval proposed changes to the Grand Gulf Nuclear Station Technical Specifications. These changes result primarily from MP&L's review of the technical specifications and surveillance procedures and are intended, in general, to enhance clarity or provide consistency with the plant design and operation. These changes have been reviewed and accepted by the Plant Safety Review Committee (PSRC) and the Safety Review Committee (SRC).

The subject changes have been evaluated by MP&L in accordance with the guidelines of 10 CFR 50.92. On the basis of that evaluation, MP&L has determined that the subject changes involve no "significant hazard consideration." Justification for this determination is provided in the attached information.

Following your review and authorization to incorporate the requested changes, MP&L will implement the affect specifications. Immediate implementation following NRC review and approval is MP&L's intention except in cases where the subject changes must await (1) the implementation of a design modification or (2) the development of implementing surveillance procedures. In general, every effort will be made to accomplish the expeditious implementation of the requested changes following NRC review and approval.

Yours truly,

LFDale

L. F. Dale
Manager of Nuclear Services

GWD/JGC/JDR:sap
Attachment

A001 1/40

cc: (See Next Page)

MISSISSIPPI POWER & LIGHT COMPANY

cc: Mr. J. B. Richard (w/o)
Mr. G. B. Taylor (w/o)
Mr. R. B. McGehee (w/o)
Mr. T. B. Conner (w/o)

Mr. J. P. O'Reilly (w/a)
Regional Administrator
Office of Inspection & Enforcement, Region II
101 Marietta Street, N.W., Suite 2900
Atlanta, Georgia 30303

Mr. R. C. DeYoung, Director (w/a)
Office of Inspection & Enforcement
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dr. Alton B. Cobb (w/a)
State Health Officer
State Board of Health
Box 1700
Jackson, Mississippi 39205

TRANSMITTAL OF PROPOSED CHANGES
TO GRAND GULF TECHNICAL SPECIFICATIONS

1. (GGNS - 96)

SUBJECT: Technical Specification 4.3.1.1 Table 4.3.1.1-1, page 3/4 3-8.

DISCUSSION: The existing specification is incorrect. Note (h) as applied to the daily channel check from APRM Flow Biased Simulated Thermal Power-High should be checking the signal which is associated with the APRM flow biasing instrumentation (i.e., drive flow).

JUSTIFICATION: Note "h", which is appended to the channel check for item 2b in Surveillance Requirement Table 4.3.1.1-1, requires that measured core flow be compared with established core flow at the existing flow control valve position. The APRM channels receive flow signals representative of driving loop flow from flow transmitters located in the recirculation loops. Drive flow, as defined in FSAR Section 7.6.1.5.6.1.1.c, is recirculation loop flow and is dependent on flow control valve position. Core flow is not measured directly but is calculated from jet pump differential pressures and represents the sum of driving and driven flow. There is a relationship between recirculation flow (drive flow) and jet pump flow (core flow), however, this relationship changes with core life. Technical Specification 4.4.1.2 verifies acceptable the relationships between core flow and drive flow every 24 hours. The terms "drive flow" and "core flow" are different and should not be used interchangeably.

Since the APRM channels receive the flow signal representative of drive flow, a comparison of measured <u>drive</u> flow with established <u>drive</u> flow at the existing flow control valve position would give a more accurate determination that the flow biased rod block/scram trip setpoint is conservative.

SIGNIFICANT HAZARDS CONSIDERATION:

The proposed change is purely an administrative change in that it corrects terminology in the APRM channel check notation. This change does not involve the reduction of safety margins. No significant increase in the probability or consequences of an accident previously evaluated is involved nor is the possibility of a new or different kind of accident from any accident previously evaluated created. Thus the proposed change to the Technical Specification does not involve any significant hazards considerations.

2. (GGNS - 399) (Resubmittal of Item #15, AECM-83/0207)

SUBJECT: Technical Specification 4.8.1.1.2.d.2, page 3/4 8-4.

DISCUSSION: Technical Specification surveillance requirement 4.8.1.1.2.d.2 should be revised so that the loads specified in this requirement are consistent with the largest single load which can be applied to the ESF busses during planned testing of the diesel generator. Surveillance requirement 4.8.1.1.2.d.2 should be revised to read as follows:

"Verifying the diesel generator capability to reject a load of greater than or equal to 1200 kW (LPCS Pump) for Diesel Generator 11, greater than or equal to 550 kW (RHR B/C Pump) for Diesel Generator 12, and greater than or equal to 2180 kW (HPCS Pump) for Diesel Generator 13 while maintaining...."

JUSTIFICATION: Regulatory Guide 1.108 section C.2.a.4 requires demonstrating "proper operation during diesel generator load shedding including a test of the loss of the largest single load.. ". The proposed revision is in compliance with Regulatory Gu .e 1.108 and specifically defines the single largest load on each ESF bus which can be verified in GGNS FSAR Tables 8.3-1, 8.3-2 and 8.3-3. The kW values given in the FSAR tables are nameplate maximum ratings. The numbers proposed for the technical specification are based on preoperational test data with the pump flows throttled to correspond to the maximum electrical load demanded by each pump. During planned surveillance testing of the diesel generators, it will not be possible to produce single loads on the ESF busses corresponding to the electrical loads included in present surveillance requirement 4.8.1.1.2.d.2.

The proposed Technical Specification change is in accordance with Regulatory Guide 1.108 and allows testing of the diesel generator response to the loss of the largest single installed load, rather than requiring a test of the loss of multiple loads greater than or equal to the nameplate rating of the largest single load.

SIGNFICANT HAZARDS CONSIDERATION:

The proposed change is purely an administrative change to the technical specifications in that it corrects the largest single load test values in accordance with the Technical Specification Bases, Regulatory Guide 1.108, and the as built plant design.

This change does not involve a reduction of safety margins and no significant increase in the probability or consequences of an accident previously evaluated is involved nor is the possibility of a new or different kind of accident from any accident previously evaluated created. Thus the proposed change to the Technical Specifications does not involve any significant hazards considerations.

3. (GGNS - 152, 211, 211a, 223, 241) (Resubmittal of Item #5, AECM-83/0254)

SUBJECT: Technical Specification 4.7.6.3.1, 4.7.6.3.2.b.1, 3.7.6.4, and 4.7.6.4; pages 3/4 7-33 and 3/4 7-34.

DISCUSSION: Technical Specification 4.7.6.3.1 requires position verification of valves in the flow path of each CO_2 system. The design of the differential pressure valves in these systems precludes position verification.

Technical Specifications 4.7.6.3.2.b.1 and 4.7.6.4.c.1 presently require destructive testing of the electro-thermal links and release of CO_2 and Halon. These Specifications should be changed to exclude actual burning of electro-thermal links and releases of CO_2 and Halon.

Technical Specifications 3.7.6.4.b and 3.7.6.4.c should be revised to specify that the actual area protected by the Halon systems is the under floor area.

Technical Specification 4.7.6.4.a is not applicable to the Grand Gulf design and should be deleted.

JUSTIFICATION: All of the CO_2 systems are supplied from a centralized CO_2 storage facility. System actuation causes repositioning of differential pressure selector valves at the storage tank and the affected area. These valves operate on differential pressure across a piston and are not designed for direct manual control. Since the valves are not designed for direct manual control, they are not provided with position indicators. There are release levers for each selector valve in the electro-manual pilot cabinets. Since mispositioning these levers could result in the opening of incorrect selector valves during a system initiation, a surveillance requirement is added. This additional surveillance maintains a commensurate level of assurance pertaining to valve position verification.

The electro-thermal links utilized for ventilation damper control are single operation devices. Once operated, they must be replaced. All of the electro-thermal links utilized in CO_2 and Halon system areas have been approved by Underwriters Laboratory. This approval is obtained through extensive testing to assure reliability of operation; destructive testing of the electro-thermal links does not provide any assurance of operability for the replacement links.

The proposed logic testing together with the additional surveillance which requires exercising of the dampers does provide assurance that the system is operable.

Technical Specifications 4.7.6.3.2.b.1 and 4.7.6.4.c.1 presently require release of CO_2 and Halon in order to satisfy the testing requirements. Since Specifications 4.7.6.3.2.b.1 and 4.7.6.4.c.2 require flow tests through the headers and nozzles, release of CO_2 and Halon during the functional testing should not be required. Automatic opening of the CO_2 differential pressure selector valves requires a CO_2 release. The testing of the system logic in 4.7.6.3.2.b.1 and the manual opening of the valves during the "Puff Test" (Specification 4.7.6.3.2.b.2) adequately verifies operability.

The Control Cabinet Room CO_2 System does not have automatic initiation capability (Reference: FSAR Table 9A-2); the Technical Specification 4.7.6.4.c.1 is revised so testing can be performed in accordance with plant design.

The Halon systems in Specifications 3.7.6.4.b and 3.7.6.4.c protect the under floor area of the PGCC. These areas are described in the FSAR, Appendix 9A, Table 9A-2, area designations OC502 and OC503. The proposed change does not affect the technical requirements.

Technical Specification 4.7.6.4.a is not applicable since the Grand Gulf Halon systems do not contain position verifiable valves in the flow path. All valves are totally enclosed, nitrogen pressure or explosive pin actuated, and can not be manually manipulated or externally verified.

SIGNIFICANT HAZARDS CONSIDERATION:

The proposed change establishes appropriate surveillance requirements for the CO_2 and Halon Fire Suppression systems consistent with the plant design. A commensurate level of safety is maintained by the proposed Technical Specification change. This change does not involve a reduction of safety margins and no significant increase in the probability or consequences of an accident previously evaluated is involved nor is the possibility of a new or different kind of accident from any accident previously evaluated created. Thus the proposed change to the Technical Specifications does not involve any significant hazards considerations.

4. (GGNS - 153, 232, 239) (Resubmittal of Item #7, AECM-83/0254)

SUBJECT: Technical Specification Table 3.7.6.6-1, page 3/4 7-39.

DISCUSSION: Technical Specification Table 3.7.6.6-1 lists the yard fire hydrants and associated hydrant hose houses which are required to be operable by Technical Specification 3.7.6.6. The elevation for these fire hydrants should be revised to the correct value. The heading title "Hydrant Number/Fire Water Loop Schedule Number" should be changed to "Hydrant Number/ Hydrant Hose House Number". The hydrant hose house numbers should be added to the column beside the hydrant numbers. The location coordinates should be corrected as noted on the markup to agree with as built locations.

JUSTIFICATION: The numbering system used in the GGNS fire protection system provides a "Hydrant Hose House Number" as a more useful identifier than a "Fire Water Loop Schedule Number"; therefore, the heading should be changed from "Hydrant Number/Fire Water Loop Schedule Number" to "Hydrant Number/Hydrant Hose House Number". The numbers in this column should also be revised accordingly. Finally, the proposed elevation for the fire hydrants corresponds to the grade elevation of 133'0". The elevation of 126'0" presently contained in Table 3.7.6.6-1 refers to the elevation of the water supply piping which is below grade elevation.

SIGNIFICANT HAZARDS CONSIDERATION:

The changes proposed to Table 3.7.6.6-1 are made to reflect actual system configuration, to correct errors and to add clarification. The changes are purely administrative in nature and do not reflect addition, deletion, or modification of any fire hydrants. This change does not involve a reduction in the margin of safety and no significant increase in the probability or consequences of an accident previously evaluated is involved nor does it create the possibility of a new or different kind of accident from any accident previously evaluated. Therefore, this change does not constitute a significant hazards consideration.

5. (GGNS - 688) (Resubmittal of Item #8, AECM-83/0254)

SUBJECT: Technical Specification 3.3.7.9 and Table 3.7.8-1, pages 3/4 3-76 and 3/4 7-43.

DISCUSSION: Technical Specification Action 3.3.7.9.a is being revised to monitor air temperature in the steam tunnel in lieu of establishing a fire watch. Table 3.7.8-1 is being revised to require area temperature monitoring in the Auxiliary Building Steam Tunnel.

JUSTIFICATION: Fire detection in the Auxiliary Building main steam tunnel (Area 1A305) is provided by two ionization smoke detectors, per Technical Specification Table 3.3.7.9-1. Establishment of the hourly fire watch required by Technical Specification 3.3.7.9.a would not be consistent with the intent of ALARA guidance since the dose rate in this area during normal operation is expected to be approximately 5 Rem/hr. The area temperature limitation for the Auxiliary Building steam tunnel is consistent with FSAR Section 9.4.6.1.2.

SIGNIFICANT HAZARDS CONSIDERATION:

The proposed change deletes the requirement to post fire watches in a high radiation area. A commensurate level of safety is maintained by requiring monitoring of air temperature in the steam tunnel. This is consistent with the allowed actions for containment and drywell areas in Technical Specification 3.3.7.9.a. This change does not involve the reduction of safety margins. No significant increase in the probability or consequences of an accident previously evaluated is involved nor is the possibility of a new or different kind of accident from any accident previously evaluated created. Thus the proposed change to the Technical Specification does not involve any significant hazards considerations.

6. (GGNS - 242) (Resubmittal of Item #15, AECM-83/0314)

SUBJECT: Technical Specification 4.7.7.2; page 3/4 7-41

DISCUSSION: The Surveillance Requirements for fire doors contained in Technical Specification Section 4.7.7.2 should be revised to agree with the requirements contained in Subsection N, Section III, of Appendix R to 10CFR Part 50.

The change incorporates the wording of the BWR/6 Standard Technical Specification with a minor modification. The words "if applicable" have been inserted to clarify that all of the required fire doors do not have automatic hold-open, release, and closing mechanisms and latches.

JUSTIFICATION: The proposed changes to the Technical Specification Surveillance Requirements assure that the requirements in Appendix R to 10CFR Part 50 related to surveillance of fire doors are correctly implemented through the surveillance requirements of 4.7.7.2.

SIGNIFICANT HAZARDS CONSIDERATION:

This proposed change documents GGNS compliance with 10CFR50 Appendix R, Section III, Subsection N with respect to surveillance of fire doors. GGNS compliance is documented in FSAR Table 9A-4. This change does not involve a significant increase in the probability or consequences of an accident previously evaluated nor does it create the possibility of a new or different kind of accident from an accident previously evaluated. Since the revised Technical Specification is in complete agreement with the appropriate requirements of 10CFR50 Appendix R, no significant reduction in the margin of safety is involved. For these reasons, this proposed change does not constitute a significant hazards consideration.

7. (GGNS - 479, 480, 486) (Resubmittal of Item #16, AECM-83/0314)

SUBJECT: Technical Specification 4.8.4.3.b, page 3/4 8-46.

DISCUSSION: Technical Specification 4.8.4.3.b establishes the setpoints and tolerances for the Reactor Protection System (RPS) electrical protection assembly (EPA) over-voltage, under-voltage, and under-frequency protective instrumentation. The bases for these setpoints is 110% of nominal for over-voltage, 97.5% of nominal for under-voltage, and 95% of nominal for under-frequency. Nominal is defined as 120 VAC at 60 hertz. With the under-voltage setpoint of 117 VAC and a 12 VAC voltage line drop from the RPS bus to the scram solenoids, a minimum voltage of 105 VAC is provided at the scram solenoids for proper operation of these devices. Although the setpoints stated in the Technical Specification are correct, the associated tolerances are not consistent with the format of the BWR/6 Standard Technical Specifications. The proposed change deletes the setpoint tolerances.

JUSTIFICATION: FSAR subsection 8.3.1.1.5.2 presents the bases for the trip setpoints. An FSAR change will be submitted to address the effect of the line voltage drop to the scram discharge coils and to reflect the EPA protective circuitry setpoint tolerances.

The General Electric design specification provides trip setpoint tolerances. The proposed change incorporates the format of the BWR/6 Standard Technical Specification and deletes the setpoint tolerances.

SIGNIFICANT HAZARDS CONSIDERATION:

This proposed change is a deletion of the setpoint tolerances for the RPS EPA protective circuitry to agree with the format of the BWR/6 Standard Technical Specification. This change does not involve a significant increase in the probability or consequences of an accident previously evaluated nor does it create the possibility of a new or different kind of an accident from any accident previously evaluated. No reduction in the margin for safety is involved. For these reasons, this proposed change does not constitute a significant hazards consideration.

8. (GGNS - 26) (Resubmittal of Item #11, AECM-83/0338)

SUBJECT: Technical Specification 3.7.6.2, 4.7.6.2.c.1.a and 4.7.6.2.c.2, page 3/4 7-31.

DISCUSSION: Additional spray/sprinkler systems should be added to the requirements for system operability of limiting condition for operation 3.7.6.2. These additional systems include systems in the Auxiliary Building, the Control Building and the Fire Pump House. The inclusion of this information in the Technical Specifications is in accordance with Inspector Follow-up Item 416/82-56-03. The words "if applicable" are added to the surveillance items which no longer apply to all of the spray/sprinkler systems listed.

JUSTIFICATION: Section 3/4 7.6 of the Technical Specification bases states that fire protection systems must be operable to ensure that adequate fire suppression capability is available to confine and extinguish fires occurring in that portion of Grand Gulf Nuclear Station (GGNS) where safety related equipment is located. Appendix 9A of the GGNS Final Safety Analysis Report (FSAR) describes the fire suppression systems which assure adequate fire protection for safety related equipment. Appendix 9A states that spray/sprinkler systems are provided to assure adequate fire protection capability in the Auxiliary Building, the Control Building and the Diesel Generator Building. The spray/sprinkler systems in the Auxiliary and Control Buildings should therefore be added to the Technical Specification Limiting Condition for Operation 3.7.6.2.

The spray/sprinkler system in the Fire Pump House serves an area which is a major element in the GGNS Fire Protection Program (although not required for safe nuclear reactor shutdown) and is therefore being added to 3.7.6.2.

The words "if applicable" are added to Specifications 4.7.6.2.c.1.a and 4.7.6.2.c.2 for the following reasons:

1) The Control Building Spray/Sprinkler Systems D154 and D155 have no automatic valves in their flow paths; therefore, Specification 4.7.6.2.c.1.a is not applicable for those systems.

2) The Spray/Sprinkler System for the Control Building, Auxiliary Building, and Fire Pump House are not dry pipe systems; therefore, Specification 4.7.6.2.c.2 is not applicable for those systems.

SIGNIFICANT HAZARDS CONSIDERATION:

The addition of surveillance requirements for spray and sprinkler systems in the Auxiliary Building, Control Building, and Fire Pump House constitute additional controls not presently included in the Technical Specifications. Therefore,

this change does not involve the reduction of safety margins. No significant increase in the probability or consequences of an accident previously evaluated is involved nor is the possibility of a new or different kind of accident from any accident previously evaluated created. Thus the proposed change to the Technical Specifications does not involve any significant hazards considerations.

9. (GGNS - X09) (Resubmittal of Item #14, AECM-83/0338)

SUBJECT: Technical Specification 6.5.2.2, page 6-9.

DISCUSSION: Section 6.5.2.2 which describes the composition of the Safety Review Committee, should be revised to reflect recent changes in corporate structure. The Assistant Vice President for Nuclear Production is now called Vice President-Nuclear, the Advisor to the Assistant Vice President, Nuclear Operations is now called the Advisor to the Vice President-Nuclear, and the Manager of System Nuclear Operations, Middle South Services, will be replaced by a designated representative of Middle South Services, Inc. who meets the experience and education requirements specified in Technical Specification 6.5.2.2.

Another change made is to the wording describing the role of consultants to Mississippi Power & Light Company to allow more than two voting consultants on the SRC.

JUSTIFICATION: The changes to the description of the SRC Chairman and indicated members are necessary to reflect changes to the Mississippi Power & Light Company corporate structure. The change in the number of voting consultants is consistent with the recommendations of the Advisory Committee on Reactor Safeguards and will allow greater use of the practical experience of these consultants.

SIGNIFICANT HAZARDS CONSIDERATION:

The change is an administrative change proposed to reflect changes to the corporate structure of Mississippi Power & Light Company and as such, corresponds to NRC example 3 (i) (10 CFR 50, Interim Final Rule, Federal Register, April 6, 1983), amendments that are not considered significant hazards considerations. Therefore, this change constitutes no significant hazards consideration.

10. (GGNS - 305, 745) (Resubmittal of Item #5, AECM-83/0356)

SUBJECT: Technical Specification 4.8.1.1.2.a.5, page 3/4 8-3.

DISCUSSION: Specification 4.8.1.1.2.a.5 presently requires load testing of diesel generators 11, 12, and 13 to greater than or equal to 50% of continuous load rating on a frequency as specified in Table 4.8.1.1.2-1 (no less often than once per 31 days). This test requires the diesel generators to synchronize onto the bus and reach 50% load in less than or equal to 60 seconds and to operate with these loads for at least 60 minutes. The proposed change to the specification would increase the loading of the diesel generators during the above testing to greater than or equal to the continuous rating.

JUSTIFICATION: In FSAR Question and Response 040.107, MP&L committed to testing in accordance with Regulatory Guide 1.108 which requires periodic testing of the diesel generators at the continuous rating. The values are being corrected to reflect this commitment.

SIGNIFICANT HAZARDS CONSIDERATION:

The proposal involves a purely administrative change which corrects present Technical Specifications on the diesel generator periodic load test requirements to comply with Regulatory Guide 1.108 and the Technical Specification Bases. There are no reductions in margin of safety. The probability or consequences of an accident previously evaluated is not increased nor is the possibility of a new or different kind of accident from any accident previously evaluated created. Thus the proposed change to the Technical Specification does not involve any significant hazards consideration.

11. (GGNS - 759) (Resubmittal of Item #13, AECM-83/0356)

SUBJECT: Technical Specification Table 4.3.7.5-1, page 3/4 3-72.

DISCUSSION: Table 4.3.7.5-1 contains accident monitoring instrumentation Surveillance Requirements. The containment/drywell area radiation monitors are required to have a CHANNEL CALIBRATION every refueling outage. The radiation monitors are required by NUREG-0737 to have a range of 1 rad/hr to 10^8 rad/hr (beta and gamma) or alternatively 1R/hr to 10^7 R/hr (gamma only). Calibration onsite by use of a radiation source is impractical due to the range of the instrument and the size of the source required. NUREG-0737, Table II. F.1-3, allows in situ calibration by electronic signal substitution for all range decades above 10 R/hr provided that calibration for at least one decade below 10 R/hr is by means of a calibrated radiation source. The proposed Technical Specification change is to add the following "*" note to the CHANNEL CALIBRATION frequency for the containment/drywell area radiation accident monitoring instrumentation.

> "* The CHANNEL CALIBRATION shall consist of an electronic calibration of the channel, not including the detector, for range decades above 10R/hr and a one point calibration check of the detector below 10R/hr with an installed or portable gamma source."

JUSTIFICATION: NUREG-0737, Table II F.1-3, states that for high-range calibration, no adequate sources exist, so an alternative was provided. This alternative is in situ calibration by electronic substitution for all range decades above 10 R/hr and calibration for at least one decade below 10 R/hr by means of a calibrated radiation source. This proposed change to the Technical Specification incorporates the NUREG-0737 provisions for high-range radiation monitor calibration.

SIGNIFICANT HAZARDS CONSIDERATION:

The proposed change specifies an alternate method for calibration of the containment/drywell area radiation monitors which is consistent with NUREG-0737. This change does not alter the intent of the Surveillance Requirement of the specification. It therefore does not significantly increase the probability or consequences of a previously evaluated accident or create the possibility of a new or different kind of accident from any accident previously evaluated. Also, it does not significantly reduce the margin of safety. Therefore, the proposed change to the Technical Specifications does not involve any significant hazards considerations.

12. (GGNS - 546) (Resubmittal of Item #14, AECM-83/0356)

SUBJECT: Technical Specification Table 4.3.7.5-1, page 3/4 3-72.

DISCUSSION: Table 4.3.7.5-1 lists the accident monitoring instrumentation surveillance requirements including requirements for the containment and drywell hydrogen concentration analyzers and monitors. The present channel calibration frequency for these monitors is quarterly. The proposed change would make the channel calibration occur on a monthly frequency.

JUSTIFICATION: The vendor for the containment and drywell concentration analyzers and monitors recommends a monthly frequency for channel calibration. This increased frequency is based on instrument drift and will ensure that instrument readings stay within drift tolerances specified by the vendor.

SIGNIFICANT HAZARDS CONSIDERATION:

The proposed change to the calibration frequency of the containment and drywell hydrogen concentration analyzers and monitors constitutes an additional limitation by increasing the calibration frequency from the present quarterly to monthly. This change does not involve the reduction of safety margins and no significant increase in the probability or consequences of an accident previously evaluated is involved nor is the possibility of a new or different kind of accident from any accident previously evaluated created. Thus the proposed changes to the Technical Specifications does not involve any significant hazards considerations.

13. (GGNS - 792) (Resubmittal of Item #2, AECM-83/0373)

SUBJECT: Technical Specification 4.1.3.1.4.a and 3.1.3.1.b, pages 3/4 1-5 and 1-4.

DISCUSSION: Specification 4.1.3.1.4.a requires the scram discharge volume to be determined OPERABLE by demonstrating the scram discharge volume drain and vent valves OPERABLE when control rods are scram tested from a normal control rod configuration of less than or equal to 50% ROD DENSITY at least once per 18 months.

Specification 4.0.4 requires the Surveillance Requirements associated with the Limiting Condition for Operation to have been performed within the required surveillance interval prior to entry into OPERATIONAL CONDITION 2. Since the referenced surveillance cannot be performed prior to entry into OPERATIONAL CONDITION 2, the following change is requested.

The proposed technical specification change adds a note to the bottom of page 3/4 1-5 to read as follows:

> "*The provisions of Specification 4.0.4 are not applicable provided the surveillance requirement is performed prior to exceeding 10% of RATED THERMAL POWER."

An exemption to 3.0.4 is also added as Technical Specification 3.1.3.1 ACTION b.3 to allow entry into Operational Conditions 1 and 2 while relying on provisions contained in the ACTION requirements.

JUSTIFICATION: Entry into OPERATIONAL CONDITIONS 1 or 2 requires the 18 month surveillance requirement be completed on the scram discharge volume as stated in Specification 4.1.3.1.4.a. However, this specification requires a normal control rod configuration of less than or equal to 50% ROD DENSITY. There are times when the plant may not be placed into OPERATIONAL CONDITIONS 1 or 2 for intervals exceeding 18 months. This change to the technical specification will allow entry into OPERATIONAL CONDITIONS 1 or 2 as long as the testing required by Specification 4.1.3.1.4.a is performed prior to exceeding 10% of RATED THERMAL POWER.

50% ROD DENSITY should be achieved at approximately 5% of RATED THERMAL POWER. The proposed Technical Specification change ensures the surveillance will be performed as soon as plant conditions permit, and gives enough margin to allow attaining 50% ROD DENSITY.

The additional 3.0.4 exemption conforms to the GE-STS (BWR/6).

SIGNIFICANT HAZARDS CONSIDERATION:

The proposed change relaxes the provisions of Specification 4.0.4 so that the plant conditions necessary to satisfy the intent of Technical Specification 4.1.3.1.4.a can be achieved. The Standard Technical Specifications in several areas indicates it is acceptable to relax the provisions of Specification 4.0.4 so that an OPERATIONAL CONDITION can be entered to achieve the plant conditions necessary to satisfy the intent of surveillance requirements. The additional 3.0.4 exemption is in accordance with the Standard Technical Specification, and is an administrative change, therefore, this change to the Technical Specification does not involve a significant reduction in a margin of safety. Additionally, as this change allows for the satisfaction of the intent of the surveillance, neither a significant increase in the probability or consequences of an accident previously evaluated nor the possibility of a new or different kind of accident from any accident previously evaluated is involved. Therefore, this proposed change to the Technical Specifications does not involve any significant hazards considerations.

14. (GGNS - 87) (Resubmittal of Item #2, AECM-83/0356)

SUBJECT: Technical Specification B3/4.6.6 and B3/4.7.2 pages B3/4 6-6 and B3/4 7-1.

DISCUSSION: Technical Specification 4.6.6.3.d.5 and 4.7.2.d.3 require verifying that the heaters in the Standby Gas Treatment System and Control Room Emergency Filtration System dissipate the proper amount when tested in accordance with ANSI N510-1975. However, ANSI N510-1975 does not address heat dissipation testing of the heaters but does address other duct heater performance tests. The proposed change adds a statement to the Technical Specification Bases to exempt the phase balance acceptance criteria (ANSI N510-1975, Section 14.2.3) from the test requirements.

JUSTIFICATION: The offsite power system consists of a non-transpositional 500 kv grid. The grid has an inherent unbalanced load distribution which results in unbalanced voltages in the plant. Voltage unbalances exceeding 5% are not atypical. ANSI N510-1975 requires current balance between phases of the heater circuits to be within 5% of one another. The intent of the ANSI requirement is to detect the heater failure by using phase balance. Due to the varying voltage unbalances at GGNS, which at times exceed the ANSI acceptance criteria, this is not a valid test. The verification of heater dissipation values to be within 10% (as required by Technical Specification 4.6.6.3.d.5 and 4.7.2.d.3, but not required by ANSI N510-1975) is adequate to verify heater operability.

SIGNIFICANT HAZARDS CONSIDERATION:

The proposed change is a purely administrative change which clarifies the Technical Specifications by exempting a portion of the ANSI N510-1975 acceptance criteria which is not applicable to the GGNS design. This change to the Technical Specifications does not involve a significant reduction in a margin of safety and it does not involve a significant increase in the probability or consequences of an accident which has been previously evaluated, nor does it create the possibility of a new or different kind of accident from any accident previously evaluated. Therefore, the proposed change does not involve any significant hazards consideration.

15. (GGNS - 828)

SUBJECT: Technical Specification Table 3.3.3-1, page 3/4 3-26; Table 4.3.3.1-1, pages 3/4 3-32 and 33; Bases Figure 3/4 3-1, page B3/4 3-7.

DISCUSSION: MP&L recently identified a concern with respect to the reactor vessel level instrumentation. Prompt notification was provided to the NRC on July 12, 1983, per Technical Specification 6.9.1.12.h. In order to evaluate the consequences of the concern an analysis has been performed. A discussion of the concern and its disposition is provided below to support the requested changes to the technical specification.

The reactor water level instrumentation at GGNS is the condensate chamber reference leg type. These instruments are strictly differential pressure devices which are reactor coolant density sensitive and are calibrated to be most accurate at the specific vessel conditions appropriate for the associated system functions. The shutdown water level range and fuel zone water level range instruments are calibrated to read accurately at atmospheric pressure; the upset, narrow and wide range water level instruments are calibrated for normal operating conditions (saturated steam at 1025 psig). At low coolant temperatures and pressures, those instruments calibrated for normal operating conditions will read higher than actual level. For example, with an actual level of 21.5" at 120°F and atmospheric pressure the narrow range instrumentation would indicate 32" and the wide range instruments would attempt to read 82.5" (in actuality the upper limit of the range of the instrument is 60").

The HPCS injection valve is interlocked closed at the vessel Level 8 setpoint (less than or equal to 55.7"). An artificially high level indication at low pressure may result in HPCS isolation when the actual vessel level is below the Level 8 setpoint. The isolation logic may be manually reset once the indicated vessel level drops below this setpoint or it will be reset automatically when the indicated vessel level reaches the Level 2 HPCS initiation setpoint. Only one accident analysis presented in FSAR Section 6.3.3, for a steam line break in the drywell, assumed that HPCS is initiated on high drywell pressure. HPCS actuation on high drywell pressure (or by manual initiation) may not result in HPCS injection due to the isolation interlock on vessel Level 8. The significance and consequences of this situation, with respect to its impact on the accident analysis, is discussed below.

Technical Specification Table 3.3.3-1.C.1 requires HPCS Drywell Pressure-High and Manual Initiation Actuation Instrumentation to be OPERABLE in various Operational Conditions. Actuation of these devices will result in vessel injection unless reactor vessel level is above the Level 8 setpoint or the Level 8 isolation has not been reset.

The proposed change to Tables 3.3.3-1 and 4.3.3.1-1 adds a note to clarify that the injection function of Drywell Pressure-High and Manual Initiation is not required to be operable during startup and shutdown conditions at such times that a false Level 8 isolation signal is present. The proposed change to Bases Figure B3/4 3-1 adds a note to clarify that the nominal level readings listed are associated with instrumentation calibrated to be most accurate at normal operating conditions. Therefore, during low pressure/temperature operating conditions, the numbers shown in the figure do not reflect actual reactor level, but nominal instrumentation setpoints. The note additionally clarifies that this situation is accounted for in the Safety Analysis.

JUSTIFICATION: The definition of OPERABILITY is general in nature and requires a subsystem/component to be "capable of performing its specified function(s)" in order to be considered operable. It is normally assumed that initiation of an ECCS system will result in vessel injection. In the GGNS design, HPCS injection is inhibited if a high reactor level condition exists. This interlock is described in section 6.3.2.3 of the GGNS Safety Evaluation Report, NUREG-0831. The interlock may be manually reset once the indicated level falls below the Level 8 setpoint permitting HPCS injection actuated either manually or by a high drywell pressure signal. The interlock is automatically reset and HPCS actuation and injection will occur at the Level 2 setpoint. Thus, the isolation logic does not make HPCS or the HPCS instrumentation incapable of performing its function but, as noted can block the injection function until Level 2 is reached. As discussed below a revised safety analysis demonstrates that HPCS initiation by high drywell pressure or manual means is not required for safe shutdown of the plant.

Based on our evaluation we conclude this concern does not present a safety hazard. The steam line break inside containment event was re-analyzed with the high drywell pressure HPCS initiation function defeated and assumed HPCS initiation on low reactor vessel level only. The model accounts for density changes in the coolant by utilizing the mass of the coolant as the parameter which actually initiates HPCS injection. Thus, the mass of coolant in the vessel is the same at the time of system initiation regardless of reactor pressure. The system response to a loss of inventory event at low pressure is essentially the same as that previously analyzed under normal operating conditions. The worst single failure for the revised case was determined to be the Division I diesel generator failure. Though there is an increase in the calculated peak cladding temperature (PCT) of about 400°F, the overall PCT of 1322°F reached is still well below the 2200°F limit of 10 CFR 50 Appendix K. In that the results of the re-analysis are less severe than the limiting LOCA analysis with regard to PCT (AECM-82/0259) and meet the requirements of

10CFR50 Appendix K, the overall conclusions of the Grand Gulf LOCA analyses are unchanged. The potential delay in HPCS initiation due to the above discussed instrumentation inaccuracies at low temperature and pressures does not adversely affect plant safety or operations.

The wide range level instrumentation initiates HPCS in a manner similar to that of the analytical model. It measures the mass of the coolant above the lower instrument tap and displays this as level. Because the instrument is not compensated for changes in water density the same mass of coolant is available in the vessel at a given indicated level (e.g., the HPCS initiation setpoint at Level 2) regardless of the pressure or actual water level.

The proposed changes make the technical specifications consistent with the plant design. Further, they clarify that the required functions of the HPCS initiation instrumentation do not always include injection and that the false indications of reactor water level due to instrumentation design and calibration requirements do not affect the safe operation of the plant.

Based on the above discussion and the re-analysis results, MP&L has provided sufficient justification for interim operation. MP&L proposes that the notes applied to Technical Specification Table 3.3.3-1 and 4.3.3.1-1 be applicable until restart following the first refueling outage. This concern will be further evaluated and design changes, if necessary, will be made prior to startup from the first refueling outage. The change to Bases Figure 3/4 3-1 is a permanent change.

SIGNIFICANT HAZARDS CONSIDERATION:

The proposed changes to the Technical Specifications represents additions of clarifying notes to reflect the as-built plant design and operability requirements of HPCS instrumentation as described in FSAR Section 7.7 and SER Section 6.3.2.3. These changes do not cause a significant increase in the probability or consequences of an accident previously evaluated nor do they create the possibility of a new or different kind of accident from any accident previously evaluated. While the analysis indicates a higher peak clad temperature the new value is not only within Appendix K limits but also well below that of the most limiting accident which was presented in AECM-82/0259, dated June 10, 1982. These changes do not constitute a significant hazards consideration.