NUREG-0831
Supplement No. 6

Safety Evaluation Report
related to the operation of Grand Gulf Nuclear Station, Units 1 and 2

Docket Nos. 50-416 and 50-417

Mississippi Power & Light Company
Middle South Energy, Inc.
South Mississippi Electric Power Association

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

August 1984

NOTICE

Availability of Reference Materials Cited in NRC Publications

Most documents cited in NRC publications will be available from one of the following sources:

1. The NRC Public Document Room, 1717 H Street, N.W.
   Washington, DC 20555
2. The NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission,
   Washington, DC 20555
3. The National Technical Information Service, Springfield, VA 22161

Although the listing that follows represents the majority of documents cited in NRC publications, it is not intended to be exhaustive.

Referenced documents available for inspection and copying for a fee from the NRC Public Document Room include NRC correspondence and internal NRC memoranda; NRC Office of Inspection and Enforcement bulletins, circulars, information notices, inspection and investigation notices; Licensee Event Reports; vendor reports and correspondence; Commission papers; and applicant and licensee documents and correspondence.

The following documents in the NUREG series are available for purchase from the NRC/GPO Sales Program: formal NRC staff and contractor reports, NRC-sponsored conference proceedings, and NRC booklets and brochures. Also available are Regulatory Guides, NRC regulations in the *Code of Federal Regulations*, and *Nuclear Regulatory Commission Issuances.*

Documents available from the National Technical Information Service include NUREG series reports and technical reports prepared by other federal agencies and reports prepared by the Atomic Energy Commission, forerunner agency to the Nuclear Regulatory Commission.

Documents available from public and special technical libraries include all open literature items, such as books, journal and periodical articles, and transactions. *Federal Register* notices, federal and state legislation, and congressional reports can usually be obtained from these libraries.

Documents such as theses, dissertations, foreign reports and translations, and non-NRC conference proceedings are available for purchase from the organization sponsoring the publication cited.

Single copies of NRC draft reports are available free, to the extent of supply, upon written request to the Division of Technical Information and Document Control, U.S. Nuclear Regulatory Commission, Washington, DC 20555.

Copies of industry codes and standards used in a substantive manner in the NRC regulatory process are maintained at the NRC Library, 7920 Norfolk Avenue, Bethesda, Maryland, and are available there for reference use by the public. Codes and standards are usually copyrighted and may be purchased from the originating organization or, if they are American National Standards, from the American National Standards Institute, 1430 Broadway, New York, NY 10018.

GPO Printed copy price: $6.50

NUREG-0831
Supplement No. 6

Safety Evaluation Report
related to the operation of Grand Gulf Nuclear Station, Units 1 and 2

Docket Nos. 50-416 and 50-417

Mississippi Power & Light Company
Middle South Energy, Inc.
South Mississippi Electric Power Association

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

August 1984

ABSTRACT

Supplement 6 to the Safety Evaluation Report for Mississippi Power & Light Company et al. joint application for licenses to operate the Grand Gulf Nuclear Station, Units 1 and 2, located on the east bank of the Mississippi River near Port Gibson in Claiborne County, Mississippi, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement reports the NRC staff's evaluation of open items from previous supplements and Technical Specification changes required before authorizing operation of Unit 1 above 5% of rated power.

CONTENTS

CONTENTS (Continued)

1 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

In September 1981, the U.S. Nuclear Regulatory Commission staff (hereinafter referred to as the Commission, NRC, or staff) issued its Safety Evaluation Report (NUREG-0831) regarding the application by the Mississippi Power & Light (MP&L) Company, Middle South Energy, Inc., and South Mississippi Electric Power Association (licensees, hereinafter collectively referred to as licensee) for licenses to operate the Grand Gulf Nuclear Station, Units 1 and 2 (hereinafter referred to as Grand Gulf), Docket Nos. 50-416 and 50-417. The Safety Evaluation Report (SER) was supplemented in December 1981 by Supplement 1, which documented the resolution of several outstanding issues in further support of the licensing activities. On June 15, 1982, the staff issued Supplement 2 to the Safety Evaluation Report (SSER 2) in which it addressed those outstanding items required to be resolved before a low-power license for Unit 1 was issued. In addition, on June 16, 1982, an operating license, NPF-13, was issued to allow Unit 1 to operate at power levels not to exceed 5% of rated power. In July 1982, the staff issued Supplement 3 to the Safety Evaluation Report (SSER 3) in which it addressed those issues remaining from previous supplements and the report of October 1981 from the Advisory Committee on Reactor Safeguards (ACRS). In May 1983, the staff issued Supplement 4 to the Safety Evaluation Report (SSER 4) that addressed primarily issues that required further evaluation before authorizing operation of Unit 1 above 5% of rated power. Supplement 5 to the Safety Evaluation Report (SSER 5) addressed the remaining issues from Supplement 4 that required further evaluation before authorizing operation of Unit 1 above 5% of rated power. This report is Supplement 6 to the Safety Evaluation Report (SSER 6) and addresses open items from previous supplements and resolution of problems in the Technical Specifications before authorizing operation of Unit 1 above 5% of rated power.

Each of the following sections of this supplement is numbered the same as the corresponding section of the SER and Supplements 1, 2, 3, 4, and 5. Each section is supplementary to and not in lieu of the discussion in the SER and Supplements 1, 2, 3, 4, and 5.

As a result of the staff's review, the following issues have been added as license conditions:

Issue	Section	License Condition
(1) TDI diesel generator reliability	8.3.1	2.C.(33)
(2) Evaluation of Technical Specification problems	16.3.1(1) (TSPS 333, 373, and 808; pages 16-37, 16-38)	2.C.(50)

In addition, the acceptability of the control room leakage rate given in Section 6.4 of this supplement is based upon the current nonoperational status of Unit 2. When Unit 2 is completed and before it is to be placed in operation, the licensee is required to provide an analysis using appropriate source terms and meteorological parameters to establish an acceptable control room leak rate for two-unit operation.

Copies of this supplement are available for public inspection at the Commission's Public Document Room at 1717 H Street, NW, Washington, D.C. and at the Hinds Jr. College, George M. McLendon Library, Raymond, Mississippi 39154. Copies of this report also are available for purchase from the sources indicated on the inside front cover.

6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.6 Containment Leakage Testing

As reported in Supplement 5 to the SER (SSER 5), the staff planned to impose a license condition to require the licensee to perform Type C leak tests (Appendix J, 10 CFR 50) of the containment isolation valves on the feedwater lines with an air or nitrogen test fluid. These leak tests were to be performed at the next scheduled outage of sufficient duration to perform the tests. However, the licensee decided to perform the pneumatic testing prior to the date when the license condition was to be issued. Consequently, the license condition is no longer required. In addition, Amendment 13 to the Grand Gulf Technical Specification includes changes to reflect the requirement that pneumatic leak testing is required for the feedwater isolation valves.

By letter dated August 13, 1984, the licensee provided partial results of pneumatic testing of containment isolation valves on the feedwater lines. On the basis of these test results, the licensee requested a schedular exemption from the requirements of 10 CFR 50, Appendix J, for acceptance criteria of leakage tests on these isolation valves. Subsequently, as indicated in its letter of August 30, 1984, the licensee reworked the seating surfaces on those valves which had excessive leakage rates during the pneumatic tests. These reworked valves were retested and found to have leakage rates within the acceptance criteria of 10 CFR 50, Appendix J. Consequently, the requested exemption cited above is no longer required. The staff concludes that this matter is now resolved.

6.4 Control Room Habitability

By letter dated June 16, 1983, the licensee submitted a revised control room habitability analysis. The staff reviewed this analysis and found it was not in compliance with the NRC guidance given in Section 6.4 of the Standard Review Plan (NUREG-0800) in that the recommended methodology with respect to atmospheric dispersion was not used and, further, the methodology that was used did not have adequate substantiation. The staff concluded that departures from the recommended methodology could be demonstrated by measuring dispersion in a wind tunnel using a physical model appropriate for the Grand Gulf facility. On August 23, 1983, the licensee elected to conduct wind tunnel tests to establish plant-specific values of effluent concentration at control room intake points. These tests were completed at Colorado State University in November 1983.

The staff has reviewed the wind tunnel test procedures and results provided by the licensee on February 6, 1984. The tests were conducted using 1:240 scale models of the nuclear station structures with two complete units including cooling towers, and with Unit 1 with its natural draft cooling tower and a partially completed Unit 2. Gaseous tracers were released from the modeled auxiliary building vents and sampled at 20 locations on the west face and at 16 locations on the top of the control building. A wind velocity profile in the vertical direction was simulated in the wind tunnel. The tests were conducted with a tunnel wind speed of 10 ft/sec at a reference height of 9.6 in., during which the tests are relatively independent of friction as represented by the Reynolds number (i.e., $\chi\bar{u}/Q$ is approximately constant when the wind speed, $\bar{u}$, and source term, Q, are varied in proportion to one another). The effects of wind direction were simulated by rotating the scaled structures on a turntable. Tests by wind direction were made for releases from the inboard and outboard vents of Units 1 and 2. $\chi\bar{u}/Q$ was calculated from measurements at each sampling location for each test. The staff concluded from its review of the test documentation and results that the experiment was well designed and that the resulting concentrations can be used for evaluation of control room habitability at Grand Gulf, Unit 1, while Unit 2 is being completed.

The wind tunnel test results provided a basis for reevaluation of atmospheric dispersion for use in the habitability evaluation of the Grand Gulf control room while Unit 1 is operating and Unit 2 is being constructed. The wind tunnel test results show that the highest relative concentrations ($\chi\bar{u}/Q$) usually occurred on the roof of the control room and that the maximum value at the rooftop sampling location was 3.5×10^{-4} m^{-2} with Unit 2 partially completed. When Unit 2 is complete, the maximum $\chi\bar{u}/Q$ value for releases from Unit 1 vents was indicated to be 4.2×10^{-4} m^{-2} for the inboard vent during west winds. Also, the maximum $\chi\bar{u}/Q$ values from Unit 2 vents was indicated to be 8.7×10^{-4} m^{-2} for the inboard vent during north-northeast winds. Thus, the relative concentration values will change when Unit 2 goes into operation. However, for purposes of this evaluation for Unit 1, $\chi\bar{u}/Q$ value of 3.5×10^{-4} m^{-2} was used as the basis because it is the highest measured value during the period of partial completion of Unit 2 and it is higher than the average rooftop value from Unit 1 vents when both units are complete and operating. This $\chi\bar{u}/Q$ value was then divided by wind speed ($\bar{u}$) at the percentiles given in the Murphy-Campe methodology (CONF-740807) for pertinent time intervals following an accident. The wind speeds at the percentiles were obtained by analysis of the onsite wind data at the 49.4 m level for the time periods August 1972 through July 1974 and the 1976 calendar year. The wind direction factors were also taken from the Murphy-Campe methodology.

The results of these analyses are as follows:

Time period	Wind direction factor	Wind speed (m/sec)	χ/Q (sec/m^3)
0 - 8 hr	1	1.2	2.9×10^{-4}
8 - 24 hr	0.88	1.6	1.9×10^{-4}
1 - 4 days	0.75	2.1	1.2×10^{-4}
4 - 30 days	0.50	3.0	5.8×10^{-5}

By letter dated February 6, 1984, the licensee requested an increase in the allowable control room leak rate from 263 cubic feet per minute (cfm) to 760 cfm. This leak rate is measured by the flow necessary to maintain a pressure differential of 1/8-in. water gauge between the control room air space and the environment. This increase is desired by the licensee to permit eventual construction of Unit 2, which will require additional cabling to be installed through the control room boundary.

The control room dose calculations described in Standard Review Plan (SRP) Section 6.4 were performed using a leak rate of 760 cfm and the atmospheric dispersion factors listed above, but with all other assumptions remaining as originally reported in the SER. The computed operator doses (0.4 rem whole body, 29.6 rem thyroid) are within the criteria of General Design Criterion (GDC) 19.

The staff finds that the control room habitability system is adequate to provide safe, habitable conditions under both normal and accident conditions without its occupants receiving radiation exposures in excess of 5 rems whole body or the equivalent to any part of the body over the duration of the accident. As a result, the staff concludes that the control room design satisfies the requirements of NUREG-0737, Item III.D.3.4, and GDC 19 and is, therefore, acceptable. This conclusion pertains to Grand Gulf Unit 1 as explained in the discussion above. At least 6 months before the operation of a second unit at Grand Gulf, the staff will review the control room habitability systems and decrease the allowable control room leak rate.

8 ELECTRIC POWER SYSTEMS

8.3 Onsite Emergency Power Systems

8.3.1 Alternating Current Power System

In support of its request for a full-power license for Grand Gulf Unit 1 and in response to an NRC Order dated May 22, 1984, the licensee submitted, by letter dated July 5, 1984, a description of the June 1984 disassembly and inspection of the Division 1 diesel generator, the postinspection engine test program, and proposed enchancements to the licensee's maintenance and surveillance program. As required by the NRC Order, the licensee submittal also addresses the similarity of the "as-manufactured quality" of the Division 1 and 2 diesel generators as part of the licensee's justification for not inspecting the Division 2 engine.

Concerns regarding the reliability of large-bore, medium-speed diesel generators of the type supplied by Transamerica Delaval, Inc. (TDI) at Grand Gulf Unit 1 and at 15 other domestic nuclear plants were first prompted by a crankshaft failure at Shoreham in August 1983. However, a broad pattern of deficiencies in critical engine components have since become evident at Shoreham, Grand Gulf Unit 1, and at other nuclear and non-nuclear facilities using TDI diesel generators. These deficiencies stem from inadequacies in design, manufacture, and QA/QC by TDI.

In response to these problems, 13 U.S. nuclear utility owners, including the licensee, formed a TDI Diesel Generator Owners Group to address operational and regulatory issues relative to diesel generator sets used for standby emergency power. The Owners Group program, which was initiated in October 1983, embodies three major efforts.

(1) Resolution of 16 known generic problem areas (Phase I program) intended by the Owners Group to serve as an interim basis for the licensing of plants.

(2) Design review of important engine components and quality revalidation of important attributes for selected engine components (Phase II program).

(3) Identification of any needed additional engine testing or inspections, based on findings stemming from the Phase I and II programs.

Pending completion of the Owners Group program, the licensee submitted a number of reports concerning its actions to ensure the reliability of the TDI diesels at Grand Gulf Unit 1. Based on its review of these reports and the status of the Owners Group program, the staff stated in its Safety Evaluation Report issued in support of the May 22, 1984, Order that additional information was needed regarding the present condition of critical engine components to support operation of Grand Gulf Unit 1 at power levels in excess of 5% of full power for the interim period pending completion of the Owners Group program and NRC staff review of recommendations stemming from this program as they apply to

Grand Gulf Unit 1. In addition to the engine inspections and subsequent post-inspection engine tests required by the Order, the staff's SER stated it would be necessary to review the licensee's proposed engine maintenance and surveillance program and any needed license conditions before issuance of a full-power license.

Appendix M to this supplement is a Technical Evaluation Report (TER) entitled "Review and Evaluation of Transamerica Delaval, Inc. Diesel Engine Reliability and Operability--Grand Gulf Nuclear Station, Unit 1." This TER was prepared by Pacific Northwest Laboratory (PNL), which is under contract to the NRC to perform technical evaluations of the TDI Owners Group generic program in addition to plant-specific evaluations relating to the reliability of TDI diesels. PNL has retained the services of several expert diesel consultants as part of its review staff.

In addition to the July 5, 1984, submittal, PNL and its consultants also reviewed the licensee submittals dated February 20, April 17, and May 6, 1984, and performed an onsite inspection of key engine components in June 1984 while the Division 1 engine was disassembled. PNL and its consultants also considered the status of the generic Owners Group program relative to the actions taken by the licensee to establish the reliability of the diesels.

Division 1 Engine

The June 1984 inspection of key engine components, including those identified by the Owners Group as known potential problem areas, indicates that these components are acceptable for nuclear service for the interim period extending to the first refueling of Grand Gulf Unit 1. This finding is subject to (1) an augmented maintenance and surveillance program and (2) operating restrictions as identified below.

Postinspection testing, as required by the May 22, 1984, Order, was satisfactorily completed. The licensee's letter dated July 2, 1984, provided the licensee's clarifications/interpretations of the required testing. Although the fast-start tests of the engine in accordance with the Order were performed subsequent to a manual prelubing of the turbocharger thrust bearings and thus did not simulate the worst challenge to the bearings, PNL does not recommend additional testing to simulate this challenge. The NRC staff concurs with this PNL finding and concludes that the tests performed by the licensee meet the intent of the NRC Order.

Division 2 Engine

In the Order dated May 22, 1984, the NRC staff stated that the need for Division 2 engine inspection would be contingent on

(1) results of the inspection of the Division 1 engine
(2) the licensee's ability to demonstrate, through a review of the manufacturer's QA records, that the two engines are of similar "as-manufactured" quality

The Division 1 engine inspection revealed only one component, the turbocharger, where failed elements, bolts and a vane, might be expected to occur in the Division 2 engine. The other components showed no rejectable indications or incipient problems that suggested adverse conditions might be present in the Division 2 engine.

Accordingly, PNL concluded that the turbochargers from the Division 2 engine should be inspected and any corrective actions taken and findings documented. No other Division 2 inspections were recommended on the basis of the Division 1 results.

In its submittal dated July 20, 1984, the licensee reported that the Division 2 turbochargers had been inspected for the type of damage found in the Division 1 turbochargers. The scope of the inspection included the stationary nozzle ring, vanes, bolts, and rotating turbine blades. The Division 2 turbochargers showed no signs of rotating disk damage, although one vane was found to be missing from each stationary nozzle ring (a similar condition was observed in the Division 1 turbochargers as discussed in Appendix M). The stationary nozzle ring bolts were found to be intact with no evidence of stress corrosion cracking. The licensee elected to replace the nozzle ring assembly and bolts although the old parts were judged to be acceptable. Turbine rotor float measurements were also performed and indicated no significant thrust bearing wear. Based on its review of the licensee's July 20, 1984, submittal, the NRC staff concludes that the licensee has satisfactorily addressed PNL's concern with respect to the Division 2 turbocharger.

On the basis of the review conducted by the licensee on the manufacturer's QA records and the upgrades accomplished for both engines, PNL concludes that the Division 1 and 2 engine components are of comparable "as-manufactured" quality. On the basis of their operating history, PNL concludes that the engines have been assembled and maintained comparably. Moreover, PNL has noted that the Division 2 engine has seen less service than the Division 1 engine. In addition, based on the status of its review of the Owners Group proposed generic resolution of the connecting rod issue, PNL has concluded that visual inspections of the connecting rods and a preload check of the connecting rod bolts should be performed on the Division 2 engine before plant operation above 5% power. In a letter dated July 26, 1984, the licensee reported that these actions have been completed. During the torque verifications, three bolts were observed to rotate from 1/16 to 1/8 inch at 95% of the required torque value. The licensee has indicated that the corresponding preload adjustment is well within the original tolerance of the torque wrench. Based on these factors and the absence of significant adverse findings from the recent inspection of Division 1 engine, the staff has concluded that no further inspections of the Division 2 engine are necessary at this time.

Augmented Maintenance and Surveillance Program

PNL concluded in the TER that modifications to the Augmented Maintenance/Surveillance Program proposed by the licensee in the July 5, 1984, submittal are needed to provide adequate assurance of engine reliability/operability. These modificatons are discussed in detail in Section 6 of the TER (Appendix M).

By letter dated July 26, 1984, the licensee committed to a revised Augmented Maintenance and Surveillance Program. The NRC staff has reviewed this letter and concludes that the licensee's program incorporates all of the modifications recommended by PNL. Therefore, the staff finds the Augmented Maintenance and Surveillance Program, as identified in the licensee's July 26, 1984, letter, to be acceptable.

Operating Restrictions

PNL recommendations and conclusions regarding TDI diesel engine reliability at Grand Gulf Unit 1 are predicated on the following assumptions:

(1) The emergency service requirements the licensee currently foresees for Grand Gulf Unit 1 will not exceed the engine load corresponding to a brake mean effective pressure (BMEP) of 185 psig. The need for this assumption is based on PNL concerns regarding the acceptability of crankshaft stresses at higher BMEP loadings.

(2) All future engine testing (except the test to obtain preturbine exhaust temperature data as described in the next section) including surveillance testing required by the plant Technical Specifications will be limited to within ± 5% of the nominal engine load where the upper limit of this load range corresponds to a BMEP of 185 psig.

(3) In the absence of the Owners Group completing all elements of their program plan, PNL's conclusions are plant specific, applying only to Grand Gulf Unit 1 and are applicable only during its first reactor refueling cycle. It is understood by PNL that at the first refueling, the licensee will implement all applicable recommendations of the Owners Group.

With regard to item (1) above, the licensee reported by letter dated July 20, 1984, that 185 psig BMEP corresponds to a generator load of 5740 kW, which is about 82% of full-rated load. This exceeds the maximum engineered safety feature (ESF) loads, 68% and 56% of full-rated load for the Division 1 and 2 engines, respectively, required to shut down the plant and maintain it in a safe condition for loss of offsite power and loss-of-coolant accident (LOCA). It also exceeds the emergency service load requirements for loss of offsite power alone which are 52% and 68% for the Division 1 and 2 engines, respectively. Thus, there exists sufficient engine capacity at 185 psig BMEP to ensure that the fuel design limits and design conditions of the reactor coolant system boundary are not exceeded, and that the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents as required by GDC 17.

The licensee also states in a letter dated August 3, 1984, that a precautionary note has been added to the Grand Gulf Off-Normal Event Procedure for Loss of Offsite Power. Specifically, the note specifies the 5740 kW (equivalent to 185 psig BMEP) limit on diesel generator loading during this off-normal event. The statement advises the operator that the Division 1 and 2 diesel generators may, if necessary, be used to carry additional plant loads. The staff found this precautionary note to be acceptable provided the "if necessary" provision would be clarified to indicate "if necessary to shut down the plant and maintain it in a safe condition." By letter dated August 5, 1984, the licensee provided acceptable clarification.

In addition, in a July 20, 1984, letter, the licensee stated that future training with respect to this procedure will explain both the basis for the note and the aspects to be taken into consideration in its application. In an August 3, 1984, letter, the licensee committed to including this training as part of the training package for Amendment 13 to the operating license which will be initiated following receipt of the amendment. The staff concurs with the need for this training and finds the licensee's proposal to be acceptable.

With regard to item (2) above, the licensee has submitted proposed Technical Specification changes incorporating this item. Specifically, the proposed changes would require that the monthly and 18-month surveillance tests be performed at a minimum of 5450 kW (78% of rated load), but not to exceed 5740 kW (82% of rated load, 185 psig BMEP). The lower limit is greater than the auto-connected loads required for the loss of offsite power and post-LOCA conditions as described above. Therefore, the staff finds these changes to be acceptable and has included them in Amendment 13 to the Technical Specifications.

With regard to item (3), the staff will include the following license condition in the full-power license amendment.

> Final evaluations and recommendations from the TDI Owners Group Program applicable to Grand Gulf Unit 1, and the licensee's actions in response to this program for the Division 1 and 2 diesel generators, shall be submitted for the NRC review and approval before plant restart from the first refueling outage.

Confirmatory Issues

PNL identified a number of confirmatory issues in Section 7.2 of the enclosed TER (reproduced as Appendix M, this report) for which PNL concluded additional information was needed to ensure that no unanticipated problems exist. By letter dated July 26, 1984, the licensee provided the requested information with the exeception of the preturbine exhaust temperature data. On the basis of its review of the July 26, 1984, submittal and discussions with its PNL consultants, the staff concludes that the information provided has not revealed any unanticipated problems and is consistent with and supportive of the PNL and staff conclusions regarding the reliability of the TDI engines at Grand Gulf Unit 1.

With regard to the preturbine exhaust temperature data, the licensee in a letter dated August 2, 1984, has committed to providing the requested data within 30 days of the date of the August 2 letter. The licensee maintains that the requested data are confirmatory in nature and not needed to ensure the satisfactory operation of the TDI engines. In support of that contention, the licensee has cited test data provided by TDI from an equivalent DSRV-16 engine for engine loads ranging to a BMEP of 225 psig. These data indicate that the turbocharger inlet temperatures are less than the turbocharger design temperature of 1200°F for BMEP loadings ranging to 225 psig. At the design Grand Gulf Unit 1 loss of offsite power/LOCA load of 158 psig BMEP, the TDI test data indicate a preturbine exhaust temperature of 1045°F. The staff concurs with the licensee's position that the requested preturbine exhaust temperature data from the Grand Gulf Unit 1 engines are confirmatory in nature and does not expect these data to alter the conclusions of this SER. Therefore, the licensee's commitment to provide the requested data within 30 days from August 2, 1984, is acceptable to the staff.

Fuel-Oil-Line Inspection

The Division 1 and 2 engines experienced several fuel-oil-line leaks between September and November 1983 which were directly attributable to minor damage to the lines from external causes, particularly from damage caused by maintenance operations. At the request of the NRC, the licensee has performed a walkdown of the Division 1 and 2 fuel-oil systems. The results of this inspection are documented in the licensee's letter dated July 26, 1984. All fuel-oil piping and tubing on the Division 1 engine were found to be in acceptable condition. However, for the Division 2 engine, a number of potential future problem areas were noted. The potential problem areas involved locations where the fuel lines were observed to be in contact with other piping or components. The licensee has completed actions to correct these potential problem areas (as confirmed by the licensee in a letter dated August 3, 1984).

As also requested by the staff, the licensee has added an additional item to its augmented maintenance and surveillance program calling for a visual inspection of the fuel-oil-piping system on both engines on a monthly basis. On the basis of its review of the licensee's July 26, 1984, submittal, the staff concludes that the licensee's actions pertaining to the fuel-oil lines are acceptable.

Conclusions

The NRC staff concludes that the TDI diesel engines at Grand Gulf Unit 1 will provide a reliable standby source of onsite power in accordance with GDC 17. This finding is based on the NRC staff/PNL review of (1) the current status of the TDI Owners Group Program in resolving the TDI diesel engine issue; (2) actions taken by the licensee to enhance and verify the reliability of the Division 1 and 2 engines, including those actions taken in response to the NRC Order dated May 22, 1984; (3) the Augmented Engine Maintenance and Surveillance Program which the licensee committed to by letter dated July 26, 1984; and (4) changes to the Technical Specifications to limit future testing of the engines to 185 psig BMEP. In addition, this finding is subject to the license condition that ensures that Grand Gulf Unit 1 will continue to meet GDC 17 beyond the first refueling outage.

9 AUXILIARY SYSTEMS

9.6 Other Auxiliary Systems

9.6.3 Emergency Diesel Engine Fuel Oil Storage and Transfer System

9.6.3.1 Emergency Diesel Engine Auxiliary Support Systems (General)

In Supplement 4 to the SER, the staff concluded that the HPCS diesel engine skid-mounted piping and components and standby diesel engine auxiliary systems (fuel oil, cooling water, air starting, and lubrication) piping and components met the requirements of GDC 2, 4, 5, and 17; met the guidance of the cited regulatory guides and standard review plans; would perform their design safety function; and met the recommendations of NUREG/CR-0660 and industry codes and standards, and are, therefore, acceptable, pending completion of the hydrostatic tests.

On the basis of the pending hydrostatic tests, the staff proposed that License Condition 2.C.(33) of Operating License NPF-13 should be revised to reflect this evaluation as follows:

(1) Paragraph (a)(2) should read: 2.C.(33)(a)(2) Provide confirmation acceptable to the NRC that HPCS diesel engine skid-mounted and standy diesel engine auxiliary systems piping has been satisfactorily tested at a minimum hydraulic pressure equal to 125% of design pressure.

By letters dated November 15, 1983, and July 28, 1984, the licensee provided confirmation that the HPCS diesel engine skid-mounted piping and standby diesel engine auxiliary systems off-engine-mounted piping were either hydrostatically or pneumatically tested in accordance with ANSI B31.1 and ASME, Section III, requirements. The staff finds this acceptable and considers this issue resolved. Therefore, the proposed license condition is no longer necessary.

9.6.7 Emergency Diesel Engine Combustion Air Intake and Exhaust Systems

In Supplement 4 to the SER, the staff concluded that the combustion air intake and exhaust system piping and components of the HPCS and standby diesel engine met the requirements of GDC 2, 4, 5, and 17; met the guidance of the cited regulatory guides and standard review plans; would perform their design safety function; and met the recommendations of NUREG/CR-0660 and industry codes and standards. They are, therefore, acceptable, subject to verification that the standby diesel engine air intake and exhaust system meets the augmented Quality Group D requirements.

On the basis of the pending verification, the staff proposed that License Condition 2.C.(33) of Operating License NPF-13 should be revised to reflect this evaluation as follows:

(2) Add (3) to paragraph (a) to read: 2.C.(33)(a)(3) Upgrade the combustion air intake and exhaust system for the standby diesel engines to meet the augmented Quality Group D requirements.

By letters dated November 15, 1983, July 28, 1984, and August 3, 1984, the licensee provided additional information and rationale for staff review and approval. The staff concurs with the licensee's rationale that the diesel generator air intake and exhaust system as installed and inspected provides an equivalent level of operational reliability and substantially the same level of safety to a system that had been designed to Quality Group C requirements, and is therefore acceptable. This issue is now resolved and, therefore, the proposed license condition is no longer necessary.

16 TECHNICAL SPECIFICATIONS

The Technical Specifications define certain features, characteristics, and conditions governing operation of a facility that cannot be changed without prior approval of the Commission. The Technical Specifications are Appendix A to the operating license. Included are sections covering definitions, safety limits, limiting safety system settings, limiting conditions for operation, surveillance requirements, design features, and administrative controls.

The Grand Gulf Unit 1 Technical Specifications were issued June 16, 1982, as Appendix A to Facility Operating License No. NPF-13. This license restricts operation to 5% power. As a result of problems associated with the implementation of surveillance requirements in the Technical Specifications, the Region II Administrator issued a Confirmation of Action letter on October 20, 1982, which required certain actions to be taken before achieving the next reactor criticality under the facility license, including submittal of license amendment requests needed to correct deficiencies in the Technical Specifications. During the next year a large number of requests for technical specification changes were submitted and acted on by the staff. Because of the large number of changes, the NRC staff performed a review of the Grand Gulf Technical Specifications. Results of this review were provided to the licensee in a meeting at the plant on January 24, 1984. The licensee had identified additional concerns that were also discussed in that meeting. NRC staff performed an inspection during February 21-24, 1984, to compare Technical Specifications with the as-built plant and found several discrepancies between the equipment identified in the Technical Specifications and equipment in the plant.

In late February 1984, after completion of NRC Region II and Office of Nuclear Reactor Regulation (NRR) inspections and reviews, it became apparent that the scope of the licensee's Technical Specification review had to be expanded to address all areas, including FSAR and as-built plant configuration. By letter dated February 24, 1984, the Director, Division of Licensing, NRR, requested that the licensee review the Technical Specifications as amended and certify that they accurately reflect the plant, the Final Safety Analysis Report (FSAR), and the Safety Evaluation Report (SER) analyses. The licensee initiated a comprehensive program to review Technical Specifications. A large task force consisting of the licensee (MP&L), Bechtel, and General Electric Company (GE) personnel was assigned responsibility in the program to ensure consistency of the Technical Specifications with the as-built plant configuration and licensing documents such as the FSAR, SER, and other supporting documents. This review was conducted under the licensee's Quality Assurance Program as the Technical Specification Review Program. Table 16.1 lists the 416 problem areas identified through the Technical Specification Review Program, gives the resolution of each problem, and references the staff's evaluation.

Table 16.1 Technical Specification Problem Sheets (TSPS)

TSPS Number	SSER 6 Section Number	TS Section (Page)	Problem	Resolution
001	N.A.	N.A.	Incorrect number of automatic depressurization system valves	TS changed by April 18, 1984 Order
002	16.4.2	3.7 (3/4 7-3)	Typographical error	TS Amendment 13
003	16.4.2	3.6 (3/4 6-54) 3.7 (3/4 7-6)	Surveillance of alarm instead of actuation channel	TS Amendment 13
004	16.4.1	3.6.3.4	No reference point for suppression pool level	TS Amendment 13
005	N.A.	N.A.	Reactor water cleanup line isolation inconsistent with as-built	TS changed by April 18, 1984 Order
006	16.4.1	3.7.4	Delete list of snubbers	TS Amendment 13
007	16.4.1	3.8.1.1	Unclear Action Statement for diesel generator surveillance	TS Amendment 13
008	16.3.2	N.A.	Change desired to allow operation with single recirculation loop	TS change not justified
009	16.4.1	3.3.6	No. of minimum operable channels for source range monitors incorrect	TS Amendment 13
010	16.4.1	3.3.7.7	Need 5 instead of 3 TIP detectors for calibration of LPRM	TS Amendment 13
011	16.4.1	3.3.6	Footnote for intermediate range monitor not needed for Grand Gulf	TS Amendment 13
012	16.4.1	3.6.3.2 3.6.3.3	Incorrect terminology and inconsistency between specifications	TS Amendment 13
013	16.4.1	3.3.2	Additional design information needed in footnote	TS Amendment 13

Table 16.1 (Continued)

TSPS Number	SSER 6 Section Number	TS Section (Page)	Problem	Resolution
014	16.3.4	3.1.3.1.4	Surveillance of SDV level instrument does not include sensors	TS change not justified
015	N.A.	N.A.	Drywell and containment pressure setpoints did not account for barometric pressure changes	TS changed by April 18, 1984 Order
016	N.A.	N.A.	Error in containment high pressure trip setpoint	TS changed by April 18, 1984 Order
017	16.4.2	3.7 (3/4 7-4)	Footnote clarification desirable	TS Amendment 13
018	16.4.2	3.3 (3/4 3-82)	Reporting clarification desirable for effluent releases	TS Amendment 13
019	16.4.2	3.6 (3/4 6-58)	Editorial	TS Amendment 13
020	16.4.1	3.6.4	Pneumatic testing of containment isolation valves	TS Amendment 13
021	N.A.	N.A.	Snubber added to list	TS changed by April 18, 1984 Order
022	16.4.1	3.3.4.1	ATWS recirculation pump trip	TS Amendment 13
023	16.4.1	3.4.2	Safety-relief valve low-low set function	TS Amendment 13
024	16.4.1	3.4.1.2	Jet pump operability	TS Amendment 13
025	16.3.3(2)	3.4.7	MSIV minimum closing time	FSAR annual update amendment
026	16.4.1	3.8.1.1	Date of diesel fuel oil sampling specification	TS Amendment 13
027	16.3.5	6.10.2	Incorrect reference to tabulation of snubbers	Superseded by TSPS 006
028	16.4.1	3.4.3.2	RCS pressure isolation valve list	TS Amendment 13

Table 16.1 (Continued)

TSPS Number	SSER 6 Section Number	TS Section (Page)	Problem	Resolution
029	16.3.2	3.3.6	Surveillance test for source range monitor	TS change not justified
030	16.3.2	3.5.1	ECCS line break instrumentation	TS change not justified
031	16.4.1	3.6.2.3	Drywell airlock seal leakage test	TS Amendment 13
032	16.4.1	3.4.3.2	RCS leakage specification	TS Amendment 13
033	N.A.	N.A.	Containment spray timer setpoints	TS changed by April 18, 1984 Order
034	16.3.5	3.3	Instrument specification review	Superseded by TSPS 346, 348, 349, 350, 351, 352, and 354 through 364
035	16.4.1	3.9.6	Refueling equipment specification	TS Amendment 13
036	16.4.1	3.12	Radiological environmental monitoring	TS Amendment 13
037	N.A.	3.2.1	Instrumentation calibration frequency	TS changed by April 18, 1984 Order
038	N.A.	3.7.1	Radiation monitor calibration frequency	TS changed by April 18, 1984 Order
039	16.3.1(1)	3.7.2	Seismic instrumentation design	First refueling TS change
040	16.4.1	3.3.2	Isolation instrumentation response times	TS Amendment 13
041	16.4.2	3.4 (3/4 4-1)	Recirculation loop operability	TS Amendment 13
042	16.3.2	3.4	Recirculation flow terminology	TS changes not justified
043	16.4.1	3.8.1.1	Diesel generator fuel-oil-pump surveillance	TS Amendment 13
044	16.3.4	3.3.3	Division 3 bus under-voltage trip time delay	TS change not justified

Table 16.1 (Continued)

TSPS Number	SSER 6 Section Number	TS Section (Page)	Problem	Resolution
045	16.4.2	3.3 (3/4 3-87)	Instrumentation radio-active gaseous effluent	TS Amendment 13
046	16.3.2	3.9.11	Change desired to allow refueling with no RHR loop operable	TS change not justified
047	16.4.1	3.3.4.2	End-of-cycle recirculation pump trip tests	TS Amendment 13
048	16.3.2	3.2.2	Change desired to extend operating domain	TS change not justified
049	16.4.1	3.2.2	Insufficient time allowed to adjust APRM gain	TS Amendment 13
050	16.4.1	3.3.7.7	Traversing incore probe surveillance	TS Amendment 13
051	16.4.2	3.1.4 (3/4 1-15)	Typographical error	TS Amendment 13
052	16.4.1	6.0	Plant manager title change	TS Amendment 13
053	16.4.2	2.1.4 (2-2)	Clarity of reactor vessel water level Action Statement	TS Amendment 13
054	N.A.	3.3.8	Containment spray actuation instruments minimum operable channels	TS changed by April 18, 1984 Order
055	16.4.1	3.4.4	Clarity of reactor coolant conductivity surveillance requirement	TS Amendment 13
056	16.3.4	3.5.1	HPCS-automatic switchover of suction	TS change not justified
057	16.4.1	3.6.1.2	Clarify surveillance interval for isolation valves	TS Amendment 13
058	16.4.1	3.7.9	Spent fuel pool temperature limit	TS Amendment 13
059	16.3.5	3.7.10	Embankment slope stability	Superseded by TSPS 133

Table 16.1 (Continued)

TSPS Number	SSER 6 Section Number	TS Section (Page)	Problem	Resolution
060	16.4.1	3.8.1.2	Clarity of diesel generator specification	TS Amendment 13
061	16.4.1	3.8.4.2	Clarity of MOV thermal overload specification	TS Amendment 13
062	16.4.2	3.6 (3/4 6-53) 3.7 (3/4 7-5)	Clarity of surveillance tests for charcoal beds	TS Amendment 13
063	16.4.1	6.0	Composition of Independent Safety Engineering Group	TS Amendment 13
064	16.3.4	6.5.1	Alternates for members of the Plant Safety Review Committee	TS change not justified
065	16.3.4	6.5.2.3	Alternates for members of the Safety Review Committee	TS change not justified
066	16.3.4	3.3.1	Variation of applicability for instruments in different tables	TS change not justified
067	16.4.1	3.6.1.2	Containment leak rate tests	TS Amendment 13
068	16.3.4	3.6.1.9	Containment purge specification clarity	TS change not justified
069	16.4.1	3.6.7.2	H_2 igniter surveillance	TS Amendment 13
070	16.4.1	3.7.6.1	Fire suppression system specification clarity	TS Amendment 13
071	16.4.1	3.7.6	Deleting special reporting requirements	TS Amendment 13
072	16.4.1	3.7.6	Visual inspection of nozzle spray heads	TS Amendment 13
073	16.4.1	3.3.7.9	Fire detecting instrumentation clarity	TS Amendment 13
074	16.4.2	3.3, 3.4, 3.8, 5.0	Footnote problems in Technical Specifications	TS Amendment 13

Table 16.1 (Continued)

TSPS Number	SSER 6 Section Number	TS Section (Page)	Problem	Resolution
075	16.4.1	3.3.3	LPCI pump discharge allowable values	TS Amendment 13
076	N.A.	3.3.3	ECCS pump response times	TS changed in April 18, 1984 Order
077	16.4.1	3.3.7.4	Remote shutdown panel	TS Amendment 13
078	N.A.	3.3.5	RCIC minimum operable channels	TS changed in April 18, 1984 Order
079	16.4.2	3.3.1 (B 3/4 3-1)	Delete reference to IEEE-279	TS Amendment 13
080	16.3.5	N.A.	Containment spray	Superseded by TSPS 054
081	16.3.1(2)	3.3.1	Reactor mode switch surveillance	Potential TS change
082	16.3.4	3.7.1	Cooling tower fan actuation testing	TS change not justified
083	16.4.1	3.3.8	Suppression pool makeup system actuation instruments	TS Amendment 13
084	16.3.1(2)	3.7.3	RCIC turbine protective trips	Potential TS change
085	16.4.1	3.11.3	Solid radioactive waste (PCP)	TS Amendment 13
086	16.4.1	3.11.1	Liquid radioactive waste update to NUREG-0473	TS Amendment 13
087	16.4.2	3.11.2 (3/4 11-11)	Gaseous radioactive waste sampling	TS Amendment 13
088	16.4.1	3.11	Radwaste reporting requirements	TS Amendment 13
089	16.4.1	3.11.1	Radioactive effluents dose calculations	TS Amendment 13
090	16.4.1	3.11.4	Radioactive effluents	TS Amendment 13

Table 16.1 (Continued)

TSPS Number	SSER 6 Section Number	TS Section (Page)	Problem	Resolution
091	16.4.2	3.12.2 (3/4 12-11)	Typographical error	TS Amendment 13
092	16.4.1	3.12.3	Radiological environmental monitoring	TS Amendment 13
093	16.4.1	6.0	Reporting requirements revised per NRC Generic Letter 84-43	TS Amendment 13
094	16.4.1	3.7.1.2	HPCS service water surveillance	TS Amendment 13
095	16.4.1	6.0	Revision of offsite organization	TS Amendment 13
096	16.4.1	6.5.2.7 (6-10)	Safety Review Committee responsibilities	TS Amendment 13
097	16.4.2	Index	Update in TS Amendment 13	TS Amendment 13
098	16.3.1(2)	3.3.7.8	Analysis may show chlorine detectors not needed	Potential TS Amendment
099	16.3.1(1)	N.A.	ADS accumulator pressure instruments	First refueling TS change
100	16.3.1(2)	3.7.8	ESF transformer life vs. temperature	Potential TS change
101	16.4.1	6.0	Update unit organization chart	TS Amendment 13
102	16.4.1	3.3.7.9	Add fire detection instruments	TS Amendment 13
103	N.A.	N.A.	Steam flow channels operable	TS changed by April 18, 1984 Order
104	16.3.2	6.1.2	Title of shift supervisor	TS change not jusitified
105	16.4.2	5.1.3 (5-1)	Effluent release boundary	TS Amendment 13
106	16.4.1	6.5.1.2	Added PSRC members	TS Amendment 13
107	16.4.2	3.6 (B 3/4 6-2)	Containment operating differential pressures	TS Amendment 13

Table 16.1 (Continued)

TSPS Number	SSER 6 Section Number	TS Section (Page)	Problem	Resolution
108	16.4.1	3.1.3.2	Scram insertion time limits	TS Amendment 13
109	16.4.2	3.3.1 (3/4 3-1)	Verification of control rod insertions	TS Amendment 13
110	16.4.1	3.3.2	RCIC flow instrumentation	TS Amendment 13
111	16.4.1	3.3.2	Radiation monitor isolation trip setpoint	TS Amendment 13
112	16.4.2	3.3.1 (3/4 3-1)	Isolation instrumentation channel/trip	TS Amendment 13
113	N.A.	3.3.2.1	Condenser vacuum instrument surveillance	TS Amendment 12
114	16.4.1	3.3.3	RCIC suction switchover to suppression pool	TS Amendment 13
115	16.4.1	3.3.3	Table footnote clarification	TS Amendment 13
116	16.4.1	3.3.3.1	Calibration of LPCS pump trip	TS Amendment 13
117	N.A.	3.3.4	Breaker interrupting time	TS Amendment 12
118	16.4.1	3.3.6	SRM rod block operability	TS Amendment 13
119	16.4.1	3.3.7.1	Dry storage monitor operability	TS Amendment 13
120	16.4.1	3.3.7.1	Offgas monitors	TS Amendment 13
121	N.A.	3.3.7.6	Count rate - SRM instrumentation	TS Amendment 12
122	16.4.1	3.3.7.12	Ventilation flow monitor tests	TS Amendment 13
123	16.4.1	3.4.1.4	RV steam temperature measurement	TS Amendment 13
124	16.4.1	3.1.1	Shutdown margin with control rod withdrawn	TS Amendment 13
125	N.A.	3.4.2.1	SRV relief setpoints	TS Amendment 12

Table 16.1 (Continued)

TSPS Number	SSER 6 Section Number	TS Section (Page)	Problem	Resolution
126	16.4.1	3.5.3.1	Suppression pool water level	TS Amendment 13
127	16.4.1	3.6.2.5	Drywell-containment differential pressure	TS Amendment 13
128	16.4.2	3.6.7 (B 3/4 6-7)	Update reference to RG 1.7, Rev. 1	TS Amendment 13
129	16.4.1	3.7.1.1	Technical Specification to allow maintenance on SSWSVS	TS Amendment 13
130	N.A.	3.7.6.1.3	Diesel fire pump battery surveillance	TS Amendment 12
131	16.4.1	3.7.6.5	Added fire hose stations	TS Amendment 13
132	16.4.1	3.7.8	Control room temperature limits	TS Amendment 13
133	16.4.1	3.7.10	Embankment stability	TS Amendment 13
134	16.4.2	3.8.1.1.2 (3/4 8-6)	Typographical error	TS Amendment 13
135	N.A.	3.8.2.1	As-built DC load profile	TS Amendment 12
136	16.4.1	3.8.4.2	Typographical error - valve numbers	TS Amendment 13
137	16.4.1	3.8.4.2.1	MOV bypass channel test	TS Amendment 13
138	16.4.1	3.11.2.1	Radioactive gas waste sampling	TS Amendment 13
139	N.A.	3.7	Corrections to snubber table	TS changed by April 18, 1984 Order
140	16.3.5	3.6.4	Isolation valve closure times	Superseded by TSPS 306
141	16.3.2	3.8.1.1.2	Test of different types of diesels	TS change not justified
142	16.3.2	3.8.1.1.2	Diesel generator (DG) cold fast starts	TS change not justified

Table 16.1 (Continued)

TSPS Number	SSER 6 Section Number	TS Section (Page)	Problem	Resolution
143	16.3.1(2)	N.A.	ESF battery DL load profile	Potential TS Amendment
144	16.4.2	3.6.1.1 (3/4 6-1)	Containment hatch seal leak tests	TS Amendment 13
145	16.4.1	3.8.1.1.2	DG test deletion generic requirement	TS Amendment 13
146	16.3.4	6.8.1	Control room limited access area	Further review showed no problem
147	16.3.5	3.3.3	AC dist. monitoring and trip	Superseded by TSPS 373
148	16.4.1	3.3.9	Tests of turbine stop and control valves	TS Amendment 13
149	16.3.4	3.3	Technical Specification instrumentation	Further review showed no problem
150	16.3.1(1)	3.3.3	ECCS instrumentation table	TS change after first refueling
151	16.3.3(1)	2.2.1	APRM setpoint discrepancy	FSAR Amendment 58
152	16.3.3(1)	3.1.1	Shutdown Δk/k margin	FSAR Amendment 58
153	16.3.1(1)	3.1.2	"Rod density" - Exxon fuel	TS change after first refueling
154	16.4.2	3.1 (3/4 1-7)	Typographical error	TS Amendment 13
155	16.4.1	3.1.3.5	Control rod position indication	TS Amendment 13
156	16.4.2	3.1.5 (3/4 1-18)	Terminology - SLC system	TS Amendment 13
157	16.4.2	3.1.5 (3/4 1-20)	Illegible figure	TS Amendment 13
158	16.4.2	3.2.2 (3/4 2-5)	APRM setpoint terminology	TS Amendment 13

Table 16.1 (Continued)

TSPS Number	SSER 6 Section Number	TS Section (Page)	Problem	Resolution
159	16.3.2	3.4.3.1	Drywell atmosphere particulate radioactivity monitoring	TS change not justified
160	16.4.1	3.4.6	RV-pressure vs. temperature	TS Amendment 13
161	16.3.1(2)	3.4.9.2 3.9.11.2	Adequacy of one ECCS loop for shutdown	Potential TS change
162	16.4.1	3.5.1	Confusing notes on RHR subsystems	TS Amendment 13
163	16.3.4	3.5.3	Clarity of specification for suppression pool	Further review showed no problem
164	16.4.2	3.6.1 (3/4 6-1)	Containment integrity definition	TS Amendment 13
165	16.3.4	3.6.1.4	Channel check of MSIV leakage control instrumentation	Further review showed no problem
166	16.3.1(1)	3.6.1.5	Feedwater leakage control system interlock	TS change after first refueling
167	16.4.2	3.6.2.1 3.6.2.3 (3/4 6-13)	Clarification of drywell integrity	TS Amendment 13
168	16.4.1	3.6.3.1	Suppression pool temperature	TS Amendment 13
169	16.4.1	3.6.3.2	Containment spray and suppression pool cooling	TS Amendment 13
170	16.4.2	3.6.4 (B 3/4 6-5)	Bases clarification	TS Amendment 13
171	16.4.2	3.6.4 (3/4 6-32)	Incorrect description of valves	TS Amendment 13
172	16.4.2	3.6.2.2 (B 3/4 6-3)	Clarify bases for drywell bypass leakage	TS Amendment 13
173	16.4.1	3.7.1.1	Clarification of SSW/ECCS	TS Amendment 13

Table 16.1 (Continued)

TSPS Number	SSER 6 Section Number	TS Section (Page)	Problem	Resolution
174	16.4.1	3.8.1.1 3.8.1.2	Clarification of usable DG fuel	TS Amendment 13
175	16.4.1	3.8.1.1	Surveillance on redundant DG	TS Amendment 13
176	16.4.1	3.8.1.2	Crane operations above upper containment pool	TS Amendment 13
177	16.4.2	3.8.2.2 (3/4 8-14)	DC sources	TS Amendment 13
178	16.3.4	3.8.4.2	MOV thermal overload	Further review showed no problem
179	16.4.1	3.8.4.2	Description of turbine valves	TS Amendment 13
180	16.4.1	3.8.4.3	RPS EPA instrument setpoints	TS Amendment 13
181	16.3.2	3.8.4.3	RPS electric power monitor surveillance	TS change not justified
182	16.4.2	3.9.1 (3/4 9-1)	Clarification of interlock terminology	TS Amendment 13
183	16.4.1	3.10.2	Rod pattern control system surveillance	TS Amendment 13
184	16.4.2	3.10.3 (3/4 10-3)	Control rod system terminology	TS Amendment 13
185	16.4.2	3.3.7.12 (3/4 3-91)	Condenser air ejector radiation monitoring	TS Amendment 13
186	16.3.4	N.A.	Followup of TMI Action items	Further review showed no problem
187	16.3.1(1)	3.7	Sampling instrument air quality	TS changed after first refueling
188	16.3.5	3.1.3 3.3	SDV vent and drain valve surveillance	Superseded by TSPS 199
189	16.3.1(2)	3.2.1 3.3.4.2	Turbine valve setpoints based on startup tests	Potential TS change

Table 16.1 (Continued)

TSPS Number	SSER 6 Section Number	TS Section (Page)	Problem	Resolution
190	16.4.1	3.12.1	Drinking water sampling and reporting	TS Amendment 13
191	16.4.1	3.11.2.1	Dose rate calculations	TS Amendment 13
192	16.4.1	3.11.2.4	Gaseous radwaste treatment - offgas system	TS Amendment 13
193	16.4.2	3.11.2.6 (3/4 11-16)	H_2 concentration measurements	TS Amendment 13
194	16.4.1	3.12.1	Food products sampling	TS Amendment 13
195	16.4.1	3.6.1.4	MSIV-LCS surveillance	TS Amendment 13
196	16.4.2	3.3.7.3 (3/4 3-64)	Meteorological system instrument operability	TS Amendment 13
197	16.4.1	3.3.6-1	Rod block instrumentation	TS Amendment 13
198	N.A.	3.3.7.1	Radiation monitor - minimum channels operable	TS changed by April 18, 1984 Order
199	16.3.4	3.3.6	Scram discharge volume level trip bypass	Further review showed no problem
200	16.3.1(2)	3.3.4	ATWS - recirculation pump trip	Potential TS change
201	16.4.1	3.3.2	Secondary containment isolation instrumentation	TS Amendment 13
202	16.3.1(2)	3.3.7.5	Accident monitoring instrumentation	Potential TS change
203	16.4.1	3.7.6.2	Control building spray/sprinkler system	TS Amendment 13
204	16.3.1(2)	3.3.2	Final setpoint change during startup program	Potential TS change
205	16.3.1(1)	3.3.3.1	High pressure core spray system operability	TS change after first refueling
206	16.4.1	3.3.3.1	LPCI/LPCS injection valve interlocks	TS Amendment 13

Table 16.1 (Continued)

TSPS Number	SSER 6 Section Number	TS Section (Page)	Problem	Resolution
207	16.3.1(2)	3.3.7.1	Radiation monitors - final setpoints	Potential TS change
208	16.3.1(1)	3.7.1.1	SSW spent fuel storage pool coolers isolation valves	TS change after first refueling
209	16.3.1(1)	3.6.5	Drywell post-LOCA vacuum breakers	TS change after first refueling
210	16.3.1(2)	3.3.1	Future installation of additional scram discharge volume level trip	Potential TS change
211	16.3.4	3.3.2	Isolation actuation instrumentation notes and radiation monitoring notes	Further review showed no problem
212	16.3.5	3.3.2	Trip channels versus trip system	Superseded by TSPS 112
213	N.A	3.3.3.1	ADS instrumentation	TS changed by April 18, 1984 Order
214	16.3.2	3.1.3.3	CRD scram accumulator surveillance	TS change not justified
215	16.3.5	3.2.2	APRM setpoints-thermal power time constant	Duplicate of TSPS 149
216	16.3.1(2)	3.3.7.5	Postaccident monitoring TMI proposed changes	Potential TS change
217	16.3.1(2)	3.3.1	Actuation logic and relay tests	Potential TS change
218	16.3.1(2)	3.3	Trip setpoint-allowable values	Potential TS change
219	16.4.1	3.4.6.1	Pressure/temperature limit curves	TS Amendment 13
220	16.3.1(2)	N.A.	Emergency lighting and portable fire extinguishers	Potential TS change

Table 16.1 (Continued)

TSPS Number	SSER 6 Section Number	TS Section (Page)	Problem	Resolution
221	16.3.2	3.4.3.2	Reactor coolant system leakage	TS change not justified
222	16.3.4	3.7.2	Control room emergency filtration system	Further review showed no problem
223	16.4.1	3.7.6.1	Add surveillance requirement to fire suppression system	TS Amendment 13
224	16.3.4	3.7.6.4	Add surveillance requirement to halon system storage tank	Further review showed no problem
225	16.4.2	5.0 (5-3) (5-4)	Illegible figures	TS Amendment 13
226	16.3.4	3.8.4.3	Clarity of RPS electric power monitoring specification	Further review showed no problem
227	16.4.1	3.8.2.1	Battery charger surveillance requirement	TS Amendment 13
228	16.4.1	3.8.4.2	Valve identifier nomenclature	TS Amendment 13
229	16.4.1	3.6	MSIV leakage control system (LCS)	TS Amendment 13
230	16.3.5	3.6.1.1	Primary containment integrity	Duplicate of TSPS 144
231	16.3.4	3.6.1.1	Containment penetrations	Further review showed no problem
232	16.3.4	3.6.4	Listing of isolation valves	Further review showed no problem
233	16.4.1	3.6.3.2	RHR flows for containment spray mode	TS Amendment 13
234	16.3.3(1)	3.3.7.5	Suppression pool water level	FSAR Amendment 58
235	16.4.1	3.6.1.3	Containment air locks	TS Amendment 13

Table 16.1 (Continued)

TSPS Number	SSER 6 Section Number	TS Section (Page)	Problem	Resolution
236	16.4.1	3.4.1.2	Jet pump operability	TS Amendment 13
237	16.4.1	3.3.6	Control rod block instrumentation setpoints	TS Amendment 13
238	16.4.2	3.3.2.3 (3/4 3-19)	Typographical error	TS Amendment 13
239	16.3.5	3.3.8	Plant system actuation instrumentation	Combined with TSPS 054
240	16.4.2	3.6.7.1 (3/4 6-56)	Containment and drywell hydrogen recombiners	TS Amendment 13
241	16.4.1	3.1.3.1	Control rod operability clarification	TS Amendment 13
242	16.3.4	3.2.1.2	APRM bases clarification	Further review showed no problem
243	16.4.1	3.4.7	MSIV stroke time definition	TS Amendment 13
244	16.4.1	3.7.6.4	Halon system valve positions	TS Amendment 13
245	16.4.1	3.7.6.2	Dry pipe sprinkler systems visual inspection	TS Amendment 13
246	16.3.4	3.7.6.2	Sprinkler systems automatic valves	Further review showed no problem
247	16.3.4	3.7.6.4	Halon system initiation	Further review showed no problem
248	16.4.2	3.11.2 (B3/4 11-4)	Misnumbered Technical Specification bases	TS Amendment 13
249	16.4.1	3.11, 3.12	Update of radiological effluent and environmental monitoring specifications	TS Amendment 13
250	16.3.4	3.1.5	Standby liquid control (SLC) applicability statement	Further review showed no problem

Table 16.1 (Continued)

TSPS Number	SSER 6 Section Number	TS Section (Page)	Problem	Resolution
251	16.4.1	3.9.2	Source range monitor count rate	TS Amendment 13
252	16.3.3(1)	5.2.1	Minimum net free air volume in containment and drywell	FSAR Amendment 58
253	16.4.1	3.3.1	Minimum number of IRM channels while shut down	TS Amendment 13
254	16.3.4	3.3.1.1	Startup channel check requirements	Further review showed no problem
255	16.4.2	3.1.3 (B 3/4 1-2)	Control rod drive bases	TS Amendment 13
256	16.4.2	3.5.1 (B 3/4 5-1)	Description in bases of HPCS performance	TS Amendment 13
257	16.4.1	3.4.2.1	Suppression pool temperature limit	TS Amendment 13
258	16.4.2	5.6.2 (5-6)	Minimum spent fuel pool level	TS Amendment 13
259	16.3.3(2)	5.7.1	Reactor vessel transient limits	FSAR annual update amendment
260	16.3.3(1)	3.6.1.8	Containment air temperature discrepancy	FSAR Amendment 58
261	16.3.4	3.1.3.3	Scram accumulator operability	Further review showed no problem
262	16.4.1	3.3.7.12	SBGT system radiation monitor	TS Amendment 13
263	16.3.4	3.3.1	Inconsistent Action Statements for RPS and EOC RPT	Further review showed no problem
264	16.3.4	3.4.7	Action Statement time limits inconsistent	Further review showed no problem
265	16.4.1	3.1.2	Exposure limits clarification	TS Amendment 13

Table 16.1 (Continued)

TSPS Number	SSER 6 Section Number	TS Section (Page)	Problem	Resolution
266	16.4.2	3.6 (3/4 6-27)	Isolation valve testing	TS Amendment 13
267	16.4.1	3.9.12	Radiation protection during spent fuel transfer	TS Amendment 13
268	16.3.1(2)	3.3.4.2.3	EOC-RPT response time and breaker circ. suppression time	Potential TS change
269	16.4.2	3.6.2.4 (3/4 6-17)	Need more specific reference	TS Amendment 13
270	16.3.4	6.8.1	Programs and procedures specification	Further review showed no problem
271	16.3.4	3.7.6.4	Halon storage require- ments	Further review showed no problem
272	16.4.1	3.4.9	Alternate shutdown cooling	TS Amendment 13
273	16.3.4	3.3.3	ADS Action Statement	Further review showed no problem
274	16.3.4	3.8.3.1	HPCS Action Statements	Further review showed no problem
275	16.4.2	3.9.8 (3/4 9-10)	Water level during refueling	TS Amendment 13
276	16.4.2	3.6.1.6 (3/4 6-9)	Need more specific reference	TS Amendment 13
277	16.4.1	3.7.7.1	Fire rated assemblies	TS Amendment 13
278	16.4.2	3.3.3 (3/4 3-25)	Need additional footnote	TS Amendment 13
279	16.3.4	3.3.3	Lack of Action Statement	Further review showed no problem
280	16.4.1	3.9.3	Friction testing in Operational Condition 5	TS Amendment 13
281	16.4.2	5.3.1 (5-5)	Fuel assemblies description	TS Amendment 13

Table 16.1 (Continued)

TSPS Number	SSER 6 Section Number	TS Section (Page)	Problem	Resolution
282	16.4.2	5.3.2 (5-5)	Difference between Technical Specification and "as-built"	TS Amendment 13
283	16.4.2	5.2.3 (5-1)	Improper building name	TS Amendment 13
284	16.3.4	3.3.7.12	Surveillance frequency inconsistent with BWR/6 Standard Technical Specifications	Further review showed no problem
285	N.A.	3.3.7.8	Chlorine detectors calibration frequency	TS changed by April 18, 1984 Order
286	16.4.2	3.7.3 (3/4 7-7)	Action Statement unclear for reactor core isolation cooling (RCIC) inoperation	TS Amendment 13
287	16.4.1	3.7.1.2	HPCS service water	TS Amendment 13
288	16.3.4	3.8.2.1	Action Statements for DC power source	Further review showed no problem
289	16.3.2	6.2.2	Neutron/monitor detector replacement	TS change not justified
290	16.4.2	6.5.2.4 (6-10)	Incorrect reference	TS Amendment 13
291	16.3.3(1)	3.3.2	Maximum MSIV isolation time	FSAR annual update amendment
292	N.A.	3.6.1.3	Air flask pressure (containment air lock)	TS changed by April 18, 1984 Order
293	N.A.	3.6.2.3	Air flask pressure (drywell air lock)	TS changed by April 18, 1984 Order
294	16.4.1	3.6.1.2	Containment leakage rates	TS Amendment 13
295	16.3.4	6.5.2.8	Audits performed under the cognizance of the SRC	Further review showed no problem
296	16.3.4	6.5.1.6	PSRC-responsibilities	Further review showed no problem

Table 16.1 (Continued)

TSPS Number	SSER 6 Section Number	TS Section (Page)	Problem	Resolution
297	16.3.4	5.4.2	Reactor vessel and recirculation system water and steam temperature discrepancy	Further review showed no problem
298	16.4.2	2.2.1 (B 2-8)	Bases for RPS setpoints	TS Amendment 13
299	16.4.2	3.7.6.3 (3/4 7-34)	CO_2 storage tank level	TS Amendment 13
300	16.3.3(1)	N.A.	MAPLHGR limits	FSAR Amendment 58
301	16.3.4	6.5.3.1	Technical review of safety-related activities	Further review showed no problem
302	16.4.1	3.8.4.1	Electrical equipment protective devices	TS Amendment 13
303	16.4.1	3.3.3	HPCS actuation instrumentation	TS Amendment 13
304	16.4.1	3.3.7.9	Fire detection instrumentation	TS Amendment 13
305	16.3.3(2)	N.A.	Potential for plant flooding from probable maximum precipitation	FSAR annual update amendment
306	16.4.1	3.6.4	Containment & drywell isolation valves	TS changed by April 18, 1984 Order and TS Amendment 13
307	16.4.1	3.9.10.2	Multiple control rod removal	TS Amendment 13
308	16.4.1	3.3.2	Setpoints for equipment area temperature - high	TS changed by April 18, 1984 Order and TS Amendment 13
309	16.4.1	3.5	Incorrect LPCI high pressure alarm setpoint	TS Amendment 13
310	16.4.1	3.5	Incorrect LPCS high pressure alarm setpoint	TS Amendment 13
311	16.4.2	3.6 (B 3/4 6-6)	Chemical release clarification	TS Amendment 13

Table 16.1 (Continued)

TSPS Number	SSER 6 Section Number	TS Section (Page)	Problem	Resolution
312	16.4.1	3.6.3.4	Fuel pool gate removal requirements	TS Amendment 13
313	16.4.2	3.1 (3/4 1-19)	Standby liquid control heaters	TS Amendment 13
314	16.3.4	3.3.1	Average power range monitor operability	Further review showed no problem
315	16.4.1	3.3.2	High steam flow trip setpoint	TS Amendment 13
316	16.3.1	3.3.2 3.3.3	High drywell pressure setpoint	Potential TS changes
317	16.3.5	3.5	HPCS pump capacity	Superseded by TSPS 256
318	16.3.5	3.1	Reactor mode switch minimum operable channels	Superseded by TSPS 197
319	16.4.2	2.0 (B 2-5)	ASME code for reactor vessel	TS Amendment 13
320	16.4.2	3.6.3 (B 3/4 6-4)	Reactor blowdown pressure	TS Amendment 13
321	16.3.2	3.6.3.1	Suppression pool limiting temperature	TS change not justified
322	16.4.2	3.5 (B 3/4 5-2)	Clarification of LPCS and LPCI bases	TS Amendment 13
323	16.4.1	3.9.2 3.10.3	SRM shorting links	TS Amendment 13
324	16.3.4	6.2.2	Shift staffing	Further review showed no problem
325	16.3.1(1)	3.6.1.9	Drywell purge valves	TS change after first refueling
326	16.3.1(2)	3.4.7.9	Spent fuel pool cooling pump room	Potential TS changes
327	16.3.3(2)	3.3.7.5	Containment/drywell area radiation monitor calibration	FSAR annual update amendment

Table 16.1 (Continued)

TSPS Number	SSER 6 Section Number	TS Section (Page)	Problem	Resolution
328	16.3.5	7.5	Containment/drywell area radiation monitor minimum channels operable	Duplicate of TSPS 367
329	N.A.	3.3.7.5	Radiation monitor applicable operating conditions	TS changed by April 18, 1984 Order
330	16.3.2	3.3.7.5	Surveillance requirements	TS change not justified
331	16.3.5	3.2	RCIC system valve leakage alarm setpoint	Considered in TSPS 032
332	16.3.2	3.5.3	Condensate storage tank water level	TS change not justified
333	16.3.1(1)	3.8.1.1	HPCS diesel generator test mode override	TS change after first refueling
334	16.4.1	3.1.4.2	Rod pattern control system	TS Amendment 13
335	16.3.4	3.8.1.1	Fuel oil sampling analysis requirements	Further review showed no problem
336	16.3.4	3.8.3.2	Action requirement inconsistency	Further review showed no problem
337	16.3.4	3.8.1.2 3.8.3.2	Clarification of DG operability requirements	Further review showed no problem
338	16.4.1	3.7.6	Fire hose stations	TS Amendment 13
339	16.3.3(1)	6.2.2	Operations personnel titles	FSAR Amendment 58
340	16.3.3(1)	6.2.2	Operations personnel rest periods	FSAR Amendment 58
341	16.3.4	6.8.2	Procedure review requirements	Further review showed no problem
342	16.4.1	3.8.1.1	HPCS diesel generator testing	TS Amendment 13
343	16.3.4	3.3.3	Battery low voltage monitor	Further review showed no problem

Table 16.1 (Continued)

TSPS Number	SSER 6 Section Number	TS Section (Page)	Problem	Resolution
344	N.A.	3.5.1.b	ECCS operability surveillance requirements	TS changed by April 18, 1984 Order
345	16.4.1	3.3.8	High reactor water level	TS Amendment 13
346	16.4.1	3.3.7.8	Chlorine detection system	TS Amendment 13
347	16.3.1(2)	N.A.	Rod pattern control system bypass switch requirements	Potential TS change
348	16.4.2	3.3 (3/4 3-63)	Meteorological monitoring instrumentation	TS Amendment 13
349	16.4.1	3.3.7.1	Isolation of fuel handling area	TS Amendment 13
350	16.4.1	3.3.2	Isolation of RHR	TS Amendment 13
351	16.4.1	3.3.7.9	Fire detection instrumentation	TS Amendment 13
352	16.4.1	3.3.7.10	Loose part monitoring system	TS Amendment 13
353	16.3.4	3.8.1.1.b.1 3.8.1.1.b.2	Diesel generator fuel storage requirements	Further review showed no problem
354	16.4.1	3.3.6	Flow-biased neutron flux rod block	TS Amendment 13
355	16.4.1	3.3.1	Surveillance frequency of RPS instrumentation	TS Amendment 13
356	16.4.1	3.3.6	Control rod block instrument surveillance	TS Amendment 13
357	16.3.1(2)	3.3.4.2	Recirculation pump trip bypass	Potential TS change
358	16.3.4	3.7.12	Radioactive gaseous effluent monitoring instrumentation	Further review showed no problem
359	16.3.4	3.8	Containment spray instrumentation	Further review showed no problem

Table 16.1 (Continued)

TSPS Number	SSER 6 Section Number	TS Section (Page)	Problem	Resolution
360	16.4.1	3.3.5	RCIC trip system clarification	TS Amendment 13
361	16.4.1	3.3.7.11	Radioactive liquid effluent monitors	TS Amendment 13
362	16.3.1(2)	3.3.2 3.3.5	RCIC time delay for actuation & isolation	TS change after first refueling
363	16.4.1	3.3.1 3.3.2	Clarity of RPS and isolation actuation design	TS Amendment 13
364	16.4.1	3.3.3	ECCS actuation instrumentation	TS Amendment 13
365	16.3.4	3.8.2.1	Battery performance service test	Further review showed no problem
366	16.3.1(2)	3.3.8	Containment spray system response time	Potential TS changes
367	16.4.1	3.3.7.5	Area radiation monitors	TS Amendment 13
368	16.3.5	3.3.7.5	Incorrect nomenclature	Considered with TSPS 329
369	16.4.1	3.3.1	RPS instrumentation specification	TS Amendment 13
370	16.3.1(2)	3.3.1	RPS trip bypass instruments not addressed in TS	Potential TS changes
371	16.4.1	3.1.3.4	Control rod drive coupling	TS Amendment 13
372	16.3.3(2)	3.2	Manual isolation of containment	FSAR annual update amendment
373	16.3.1(1)	3.3	Division 3 undervoltage protection	TS change after first refueling
374	16.3.1(2)	3.7.5	Postaccident monitoring instrumentation	Potential TS changes
375	16.4.1	3.2.3	MCPR Bases references	TS Amendment 13

Table 16.1 (Continued)

TSPS Number	SSER 6 Section Number	TS Section (Page)	Problem	Resolution
376	16.3.3(2)	3.6.6.1	SBGT flow limit	FSAR annual update amendment
377	16.3.1(2)	3.7.2	Control room in-leakage	Potential TS changes
378	16.3.4	3.6.6.1 3.6.6.3	SBGT surveillance inconsistency	Further review showed no problem
379	16.4.2	3.6 (B 3/4 6-1)	Air lock seal decay test	TS Amendment 13
380	16.3.2	3.10.1	Low power physics test	TS change not justified
381	16.3.4	3.3.1	Instrumentation terminology	Further review showed no problem
382	16.4.1	3.4.1.1	Hydraulic instability	TS Amendment 13
800	16.3.3(1)	3.2.2	APRM flow biased scram setpoint	FSAR Amendment 58
801	16.3.3(1)	3.6.1.7	Containment pressure	FSAR Amendment 58
802	16.3.3(1)	3.3.4.2	Recirculation pump trip	FSAR Amendment 58
803	16.3.4	3.3.7.1	Radiation monitors	Further review showed no problem
804	16.3.4	3.8.1.1	DG fuel oil tank capacity	Further review showed no problem
805	16.3.2	3.1.5	Sodium pentaborate volume	Changes not justified
806	16.3.3(1)	3.2.3	MCPR limit	FSAR Amendment 58
807	16.3.3(2)	N.A.	Radiation monitor Surveillance Requirements	FSAR annual update amendment
808	16.3.1(1)	3.8.1.1.2	DG protective trip design	TS changes after first refueling
809	16.3.4	3.8.4.2	MOV thermal overload bypass circuitry	Further review showed no problem
810	16.3.3(2)	3.8.4.1	Trip setpoints for circuit breakers	FSAR annual update amendment

Table 16.1 (Continued)

TSPS Number	SSER 6 Section Number	TS Section (Page)	Problem	Resolution
811	16.3.4	6.5.1	Plant Safety Review Committee requirements	Further reviewed showed no problem
812	16.3.3(1)	3.3	Main steam tunnel temperature timer	FSAR Amendment 58
813	16.3.4	6.5.2.2	Manager of QA requirements	Further review showed no problem
814	16.3.4	6.5.2.2	Safety Review Committee requirements	Further review showed no problem
815	16.3.4	3.4.4	Reactor coolant chemistry	Further reviewed showed no problem
816	16.3.3(1)	3.2	Main steam line flow-high instrumentation	FSAR Amendment 58
817	16.3.3(1)	N.A.	Standby gas treatment system capacity	FSAR Amendment 58
818	16.4.2	3.6 (3/4 6-46)	Secondary containment isolation terminology	TS Amendment 13
819	16.3.3(1)	N.A.	Seismic instrumentation nomenclature	FSAR Amendment 58
820	16.3.3(1)	3.2	RCIC instrument setpoints	FSAR Amendment 58
821	16.3.4	3.1.3.3	Control rod drive accumulator level	Further review showed no problem
822	16.3.3(1)	3.6.6.3	Standby gas treatment system performance	FSAR Amendment 58
823	16.3.3(1)	6.2	Auxiliary building isolation dampers	FSAR Amendment 58
824	16.3.4	6.4	Containment and drywell isolation valves	Further review showed no problem
825	16.3.3(2)	3.2	RCIC isolation instrumentation	FSAR annual update amendment
826	16.4.1	3.7.9	Spent fuel pool cooling pump operability	TS Amendment 13

Table 16.1 (Continued)

TSPS Number	SSER 6 Section Number	TS Section (Page)	Problem	Resolution
827	16.3.4	3.7.6.1	Fire suppression water system pressure	Further review showed no problem
828	16.3.3(1)	3.3.5	RCIC actuation instrumentation	FSAR Amendment 58
829	16.3.4	3.4.4	Reactor coolant system chemistry requirements	Further review showed no problem
830	16.3.3(1)	3.4.1.4	Reactor vessel thermal cycle limit	FSAR Amendment 58
831	16.3.3(1)	3.1	RPS response times	FSAR Amendment 58
832	16.3.3(1)	3.6.6.3	SGTS logic description	FSAR Amendment 58
833	16.3.3(1)	3.8.1.1	Diesel generator load reject capability	FSAR Amendment 58

16.1 Licensee's Technical Specification Review Program

The licensee's Technical Specification Review Program (TSRP) was presented to the NRC staff in a meeting at Bethesda, Maryland, on March 9, 1984 and documented by licensee's letter dated March 18, 1984. The stated objectives of this review were to (1) verify the consistency of the Technical Specifications with the as-built plant configuration, the FSAR and the staff's SER; (2) identify, submit, and support issuance of necessary changes to the Technical Specifications in a timely manner; and (3) provide the necessary assurance that the Technical Specifications are accurate and adequate. The staff reviewed the licensee's program as described in the March 18, 1984, letter. The staff also reviewed the licensee's implementation of its program during a site visit on March 28-30, 1984. The site visit was performed by NRR and Region II personnel.

Acceptance Criteria for TSRP

To determine the adequacy of the licensee's Technical Specification Review Program, the staff established the following acceptance criteria and reviewed the program against these criteria.

(1) The program should cover the entire scope of the Technical Specifications and should consider the adequacy of that scope.

(2) The program should require comparison of the Technical Specifications with the various licensing documentation (FSAR, SER, etc.) and with the as-built plant.

(3) The program should include appropriate management involvement and oversight by the licensee.

(4) The program should require substantial licensee involvement in all appropriate aspects of the program.

(5) The program should involve all appropriate parties including GE, Bechtel, and appropriate representatives from the licensee.

(6) The program should be under the licensee's Quality Assurance program.

Evaluation of TSRP

The licensee divided the Technical Specifications into four portions. These four portions were (1) the Technical Specifications and associated bases within the scope of the nuclear steam supply system (NSSS) vendor (GE); (2) the Technical Specifications and associated bases within the scope of the architect-engineer (Bechtel); (3) the Technical Specifications and associated bases concerning Radiological Effluent Technical Specifications (RETS); and (4) the Technical Specifications concerning definitions, design features, and administrative controls. This division of the Technical Specifications and their subsequent assignment to appropriate review groups ensured that all the Technical Specifications were reviewed by at least one, and in some cases, more than one review group. Therefore, the staff concluded that the program covered the entire scope of the Technical Specifications.

The program required that the Technical Specifications be used as a focal point for comparing them with the FSAR, SER, BWR/6 Standard Technical Specifications (STS), and the as-built plant. The staff found this comparison requirement desirable and acceptable; however, the staff believed that there was an inherent deficiency in the program in that by establishing the Technical Specifications as the focal point of the review, the program presumed the Technical Specifications to be sufficient in scope and in mode applicability. The staff believed that in addition to comparing the Technical Specifications with the BWR/6-STS, the program should require consideration of the Technical Specifications to determine if their scope and mode applicability are adequate and if the Technical Specifications contain unnecessary requirements. The staff's reasons for this concern were that the BWR/6-STS were prepared in conjunction with the Technical Specifications and their scope and mode applicability have not been proven through use on other plants as have the other Standard Technical Specifications currently in use. Furthermore, the BWR/6-STS have not been endorsed by the staff. Therefore, the BWR/6-STS are not considered to provide the sole basis for determining that the Technical Specifications are adequate in scope or mode applicability. The staff believed that the program should contain elements requiring the reviewers to consider whether the Technical Specifications contain the appropriate scope and mode applicability requirements and that the requirements should be verified by comparing the Technical Specifications with the assumptions used in the Grand Gulf safety analyses. By letter dated July 5, 1984, the licensee demonstrated that, although the BWR/6-STS were used during the Technical Specification Review Program, they were not solely relied upon to justify the acceptability of the Technical Specifications. The licensee's response also indicated that the Technical Specification Review Program contained review elements and mechanisms that resulted in consideration of operating mode applicability. The NRC staff concludes that the scope and mode applicability requirements have been adequately considered.

The staff also noted that although the program provided for a direct comparison of the Technical Specifications with the as-built plant and with the FSAR, it did not provide for a direct comparison of the as-built plant to the FSAR. In view of the fact that a number of discrepancies between the FSAR and the as-built plant were detected during other recent such comparisons (e.g., during the Region II inspection of February 21-24, 1984, see NRC Inspection Report 416/84-06), the staff believed that such a comparison would have provided added assurance that the as-built plant is accurately described in the FSAR. However, a limited comparison of the as-built plant to the FSAR was possible through cross comparisons of the as-built plant to the Technical Specifications and the FSAR to the Technical Specifications. The staff believes this limited comparison is acceptable.

The licensee established a project-oriented organization reporting to the Senior Vice President, Nuclear, to coordinate the review effort. The Project Manager, Technical Specifications Review, is an MP&L employee who reported directly to the Senior Vice President, Nuclear. The Review, Prioritization and Direction (RPD) Manager, Administrative Manager, Radiological Effluent Technical Specification (RETS) Manager, and the NSSS/BOP Manager were all MP&L employees and they reported directly to the Project Manager. The staff concludes that the program provided for appropriate MP&L management involvement and oversight.

The RPD Manager is the Plant Staff Technical Superintendent. The RPD Group included representatives from Nuclear Plant Engineering (NPE), Plant Staff and Nuclear Safety and Compliance. The primary functions of the RPD Group were to (1) evaluate findings forwarded to it, (2) assign priority to potential changes to the Technical Specifications, (3) direct necessary corrective action, and (4) concur with findings or adequacy of completed or proposed corrective actions. The NSSS/BOP Manager had an NSSS Manager and a BOP Manager, both of whom were MP&L employees, reporting to him. Also reporting to the NSSS/BOP Manager was an Onsite Review Team whose minimum composition included (1) a GE or Bechtel engineer, (2) an MP&L NPE engineer, and (3) an MP&L Senior Reactor Operator. The initial review of Technical Specifications within the NSSS/BOP Review Group's scope of responsibility was conducted in the GE and Bechtel home offices, as appropriate, followed by some field verification at the plant site. The Administrative Review Group was directed by an engineer from the MP&L Quality Assurance organization. The RETS Review Group Manager is the MP&L Manager of Radiological and Environmental Services. The staff's review showed substantial licensee involvement in all appropriate aspects of the program, which is considered preferable to the delegation of such involvement to a consultant or other organizations. The staff's review also showed that the program required the involvement of all the appropriate parties in this review effort.

The program provided for auditing its implementation and effectiveness by the licensee's Quality Assurance organization. A Quality Engineer was assigned to the program to provide quality control support to the Project Manager.

Therefore, the staff concluded that the program included commitments for adequate participation by the licensee's Quality Assurance organization.

The staff conducted an onsite inspection on March 28-30, 1984, to review the licensee's implementation of its Technical Specification review program. The staff interviewed several members of the licensee's organization as well as several GE and Bechtel representatives, and examined several Technical Specification review packages during its onsite review. From its examination of these packages, the staff determined that the licensee implemented the program as described in its submittal of March 18, 1984. However, the staff's review also disclosed that the licensee (particularly in the Bechtel areas of review responsibility) was apparently using the draft BWR/6-STS as justification for the acceptability of the Technical Specifications. For reasons given above in this section, the staff indicated that although the BWR/6-STS can serve as a useful guide in evaluating the adequacy of the Technical Specifications, a determination regarding the acceptability of the Technical Specifications based soley on a comparison with the BWR/6-STS is inadequate. The licensee stated that in the final closure of Technical Specification packages the draft BWR/6-STS would not be used as sole justification for determining the acceptability of the Technical Specifications but that additional justifications would be provided.

By letter dated July 5, 1984, the licensee responded to NRC staff concerns regarding overreliance on the BWR/6-STS for justification of the acceptability of Grand Gulf Technical Specifications. Results of the review program showed that about 20% of the inconsistencies were found by comparison with the BWR/6-STS, whereas the remainder of the inconsistencies were found by comparison with the as-built plant, FSAR, SER, and design documents. However, the BWR/6-STS were not considered to be sufficient justification for changing the Technical Specifications. All changes were required to be based upon engineering and licensing requirements. The staff concludes that the Technical Specification Review Program provided adequate bases for determining acceptability of the Technical Specifications.

The staff observed that potential Technical Specification problems could be identified during reviews by various reviewers and determined to be insignificant by the RPD Manager. When such a determination was made, item numbers were not assigned to the Technical Specification Problem Sheets and therefore items that were actually significant were possibly not identified and hence dropped from further consideration. The staff believed that this was a deficiency in the program and that a tracking system for such items should be developed and implemented. The licensee informally committed to implement a tracking system for such items. By letter dated July 5, 1984, the licensee provided a description of the review process to show that reviewer comments were appropriately considered and dispositioned. The NRC staff concludes that all problems were appropriately considered.

The staff noted that the program did not specifically require a search of the FSAR for additional items that are plant specific and are not presently addressed in the Technical Specifications. The licensee informally committed to a followup verification program to address this issue. By letter dated July 5, 1984, the licensee responded to this concern, by describing the review procedure which required the reviewer to examine the FSAR and supporting documents, the SER, and as-built documentation to ensure that Grand Gulf Unit 1 unique features were properly addressed in the Technical Specifications. The NRC staff concludes that plant-specific features were adequately addressed.

Although not required by the staff, the licensee selected the Impell Corporation to perform an independent audit. The Impell audit was in progress during the NRC staff onsite inspection. Impell had selected and was in the process of reviewing 11 features of the Grand Gulf design to determine if these features were adequately covered in the Technical Specifications. Impell concentrated its selection of features to be reviewed on features unique to the BWR/6 and Mark III containment designs, and to the Grand Gulf plant-specific design. The staff received a briefing on the Impell review on April 4, 1984, and concluded that the 11 features selected for review were an adequate sample to determine if these unique features were covered in the Technical Specifications.

A final report prepared by the Impell Corporation of its review of the effectiveness of the MP&L TSRP was transmitted to NRC by letter dated April 16, 1984. The Impell Corporation conclusions follow:

(1) The Technical Specification Review Program process and results provide adequate assurance that the Technical Specifications, as revised to reflect the results of Impell's review, accurately reflect the Grand Gulf design analyses and the as-built plant. Although the possibility remains that undiscovered Technical Specification discrepancies may still exist, it is unlikely that such discrepancies would be of substantial safety significance.

(2) The Technical Specification Review Program process and results provide adequate assurance that the Technical Specifications, as revised to reflect the results of the Impell's review, appropriately reflect the unique design features of the Grand Gulf Nuclear Station. Impell believes that it would be prudent to confirm further the Technical Specification coverage of plant-unique features and notes that the licensee has initiated such a confirmatory program.

(3) The Technical Specification Review Program process and results provide adequate assurance that the Technical Specifications, as revised to reflect the results of Impell's review, meet or exceed current NRC/ industry standards for the level of detail to be included in Technical Specifications.

(4) The Technical Specifications, as revised to reflect the results of Impell's review, will be adequate to ensure safe operation of the plant.

The staff has reviewed Impell's report and concurs with its conclusions regarding the TSRP process and its implementation. The licensee by letter dated July 5, 1984, provided the results of a confirmatory program of the review of Technical Specifications for the features unique to BWR/6 and Mark III designs. The licensee found no Grand Gulf BWR/6 Mark III unique design features that were not already addressed in the Technical Specifications. Based on its review of the review results, the staff concludes that unique features were adequately considered.

In response to the staff's request, the licensee asked the General Electric Company (GE) to perform an overview review of the portions of the Grand Gulf Technical Specification Review Program that were completed by the licensee and

Bechtel. The completion and results of this GE review were described in a letter from the licensee dated June 8, 1984. The licensee characterized the GE findings as enhancements or, in some cases, improvements in wording. The successful completion of the GE review provides additional assurance that the Technical Specification Review Program was adequate.

The staff has concluded that there is reasonable assurance that the licensee's program as implemented has accomplished its intended objectives.

16.2 Results of the Licensee's TSRP

The licensee developed a system of tracking problems found in the Technical Specifications and their resolution as a part of the Technical Specification Review Program. Technical Specification Problem Sheets (TSPS) were prepared for each problem giving the problem description, anticipated resolution, and disposition. All previously identified problems, as well as those identified during the TSRP, were processed in accordance with TSRP procedures. A total of 416 TSPS resulted from the program. This total includes Technical Specification changes previously requested by the licensee, and NRC staff comments on Technical Specifications resulting from staff reviews and inspections.

The results of the licensee's Technical Specification Review Program were presented to the NRC staff in a meeting on April 4, 1984, and documented by letters dated April 9, April 19, May 1, May 8, and July 5, 1984. During the course of the licensee's review, the staff met with the licensee on March 14, March 22, March 28, April 4, April 5, April 11, April 20, and April 27, 1984, to discuss potential Technical Specification changes and to provide comments resulting from its review and inspections of Grand Gulf Technical Specifications. Letters requesting additional information were sent to the licensee on September 12, 1983, and March 19, April 20, April 25, and May 8, 1984. Licensee's responses to NRC's staff requests were sent in letters dated September 12, 1983, April 17, April 26, April 30, May 8, May 24, May 25, May 30, June 20, July 3, and August 2, 1984.

Based on its review of Technical Specification review problem areas, the licensee determined that there were 23 problem areas that were nonconservative with respect to the safety analyses in the FSAR and in the SER. Accordingly, requests for changes to the Technical Specifications were submitted by letters dated March 20, March 29, April 7, April 10, and April 11, 1984.

The NRC staff also reviewed the problem areas identified by the licensee's TSRP, which included problem areas found by staff's comparison of Technical Specifications with the FSAR and the SER. For operation under the low power license (5% power) the staff found that 22 of the 23 changes requested by the licensee would be in the direction of increased safety and should be made for operation under the low power license. The remaining change could have permitted unmonitored release of radioactive gas and therefore was not ordered.

An Order was issued on April 18, 1984, to amend the Technical Specifications in these 22 problem areas.

By letter dated May 9, 1984, the NRC staff advised the licensee that all changes to Technical Specifications identified on its Technical Specification Problem Sheets must be considered for full-power operation. The licensee was requested to provide a marked-up copy of the effective Technical Specifications to show

how problems would be resolved by changes. During May and part of June 1984, the NRC staff met with the licensee to determine those problem areas that must be resolved by changes to Technical Specifications before full-power operation and appropriate wording of changes to the Technical Specifications.

By letters dated June 17, 18, 19, 20, 21, and 22, and August 14, 1984, the licensee submitted proposed changes to the Technical Specifications determined by the NRC staff to be required for full-power operation. These letters also provided a description of the change and justification for the change.

The staff review of the 416 MP&L Technical Specification Problem Sheets and the MP&L letters describing and justifying changes to the Technical Specifications was performed by various branches in the Office of Nuclear Reactor Regulation. Evaluations of problem areas that do not require a Technical Specification change for authorization of full-power operation and evaluations of proposed changes to the Technical Specifications are provided in Sections 16.3 and 16.4, respectively, of this supplement. Table 16.1 lists the 416 problem areas, gives the resolution of each problem, and references the staff's evaluation. Changes needed for a full-power license will be issued in Amendment 13 to the operating license after full-power operation is authorized by the Commission.

16.3 Evaluation of the Licensee's Technical Specification Problem Sheets

The staff reviewed the 416 Technical Specification Problem Sheets (TSPS) resulting from licensee's review of the Grand Gulf Technical Specifications. Those problems the staff determined should be resolved by changes to the Technical Specifications before the initial power escalation above 5% power are considered in Section 16.4 of this supplement. Those problems the staff determined should be resolved by means other than Technical Specification changes before initial power escalation are considered below in five subsections grouped according to reasons indicating why no changes to Technical Specifications are required.

16.3.1 Future Technical Specification Changes

(1) Technical Specification Changes Required at the First Refueling

TSPS 039 Seismic Instruments

This problem concerned the four pipe-mounted triaxial peak recording accelerographs used for post-seismic event evaluation. The range of these instruments is too low to prevent damage by overranging during normal pipe movement and vibration. These instruments, together with other seismic instruments, provide information for evaluating potential damage following a seismic event. The licensee is evaluating a design change to provide improved post-seismic event evaluation and will submit Technical Specification changes on a schedule consistent with implementation of required design changes before restart following the first refueling outage. The Technical Specification requires that a report be submitted describing corrective action if seismic instruments became inoperable. This report has been submitted for these four instruments. The staff concludes that no change to the Technical Specification is required at this time because the instruments are not used to actuate engineered safety features and post-seismic damage could be evaluated, if necessary, using conservative analyses based on the other operable seismic instruments.

TSPS 099 Automatic Depressurization System Accummulators

By letters dated October 26, November 9, 1982, and October 24, 1983, the licensee committed to install additional instrumentation to monitor the automatic depressurization system (ADS) air receiver pressure by the end of the first refueling outage. At that time, changes to the Technical Specifications will be requested to include the applicable operability requirements, action statements, and surveillance requirements. Because the licensee is not required to complete installation until the first refueling, changes to the Technical Specifications are not required now.

TSPS 150 Automatic Depressurization System Logic Modifications

License Condition 2.C.(44)(i)(a) requires installation of improved ADS logic instrumentation before startup following the first refueling. Technical Specifications are not required for these items until they are installed in the plant. Because the licensee is not required to complete installation until the first refueling outage, changes are not required for a full-power license amendment.

TSPS 153 Use of Exxon Fuel for Reload

This problem concerned the use of the term "Rod Density" in Technical Specifications 3.1.2, which requires comparison of predicted and measured rod densities to detect possible reactivity anomalies. This term may not be appropriate for Fuel Cycle 2 and beyond since Exxon fuel will be loaded for those cycles.

This item is not relevant to Fuel Cycle 1 operation and therefore no change to the Technical Specifications is required.

TSPS 166 Feedwater Leakage Control System Interlocks

This problem concerned the ability of the feedwater leakage control system (FWLCS) to maintain a water seal in the feedwater system following a LOCA. The NRC staff requires that feedwater valves F010 A and F010 B, F032 A and F032 B, and F065 A and F065 B be pneumatically leak tested in accordance with Appendix J of 10 CFR 50, unless analyses can demonstrate that the FWLCS can maintain a water seal following a LOCA. Technical Specification 4.6.1.5 does not currently contain surveillance requirements for the referenced valves.

The full-power license amendment will include a license condition requiring that (1) isolation valves F010 A, F010 B, F032 A, F032 B, F030 A, F030 B, F063 A, F063 B, and F065 A, F065 B shall be "Type C" leak tested according to Appendix J to 10 CFR 50; and (2) before startup following the first refueling outage, the licensee shall either demonstrate why the containment isolation valves mentioned above should not be "Type C" leak tested or integrate these isolation valves into the "Type C" Surveillance Requirements of the Technical Specifications. Therefore, the staff finds that a Technical Specifications change at this time is inappropriate.

TSPS 187 Air Supply for Safety-Related Components

By letter dated November 9, 1982, the licensee has committed to perform annual sampling of instrument air supplied to safety-related components for air quality beginning at the first refueling. The current Technical Specifications do not include requirements for instrument air sampling. Based on its review, the staff concludes that since the instrument air system was tested before operation, it is acceptable to have the first quality test performed at the first refueling. Thus, it is appropriate to not add a Technical Specification for air sampling until the first refueling.

TSPS 205 High-Pressure Core Spray System Operability

This problem sheet identified a potential problem regarding a footnote that permits the high-pressure core spray system to be inoperable under certain conditions until startup following the first refueling outage. From an additional review of the Technical Specification requirements, the licensee has determined that no change is required at this time. A Technical Specification change will be required before startup following the first refueling outage to either revise the Technical Specifications so the requirements remain applicable for subsequent plant operations, or delete the note following a system design modification. Based on its review of the problem sheet and discussions with the licensee, the staff finds that no Technical Specification changes are required at this time. Technical Specification changes will be required before startup following the first refueling outage.

TSPS 208 Standby Service Water System Design

The standby service water system (SSWS) does not have sufficient capacity under the design-basis-accident heat load conditions to remove heat from the spent fuel pool. Thus the current Technical Specification requires verification that the valves isolating the standby service water from the spent fuel pool cooler are locked closed. License Condition 2.C.(28) requires the licensee to design and install increased capacity for the SSWS to include heat removal from the spent fuel pool at the first refueling outage.

This problem sheet was initiated to delete the Technical Specification requirement to verify that isolation valves are locked closed. However, this change is not appropriate until increased SSWS capacity is installed and tested during the first refueling outage.

TSPS 209 Drywell Vacuum Breaker Design

The Technical Specifications regarding drywell post-LOCA vacuum breaker operability contains a note that modifies an Action Statement and a Surveillance Requirement until restart after the first refueling outage. This note applies to the position indication verification of the vaccum breaker isolation valves. A design change to the vacuum breakers will be incorporated during the first refueling outage, at which time the note will be changed. This problem sheet was initiated to indicate that Technical Specifications change will be necessary when a design change is implemented at the first refueling outage. Therefore, a change is inappropriate at this time.

TSPS 325 Containment Purging

License Condition 2.C.(19) requires the drywell purge valves to be sealed closed when operating in Modes 1 through 3 until the drywell purge valves are qualified. This problem sheet addresses one part of the license condition that restricts the time for containment/drywell purging. The question raised was whether to convert the license condition to a Technical Specification. As reported in SSER 5, the staff planned to impose a two-section license condition; one section would pertain to containment purging and the other section to drywell purging. Subsequently, the staff decided to put the drywell purging section in the Technical Specifications. The containment purging section will remain as a license condition during the first fuel cycle. The replacement of the license condition with Technical Specifications will be reevaluated at the first refueling outage.

TSPS 333 HPCS Diesel Generator Test Mode Design

The staff's Safety Evaluation Report Section 8.3.1 requires that the diesel generator unit design should include an emergency override of the test mode to permit response to genuine emergency signals and to return the control of the diesel generator to the emergency standby mode. The licensee did not provi this design feature on the HPCS diesel generator.

By letters dated July 3 and August 2, 1984, the licensee committed to implement this design feature for the HPCS diesel generator at the first refueling outage The staff concludes that the present design feature would not affect reliability of the HPCS diesel generator for this short period of time. The full-power license amendment will include a condition requiring the incorporation of this design feature before startup after the first refueling outage.

The present Technical Specification is in accordance with the as-built design and, therefore, no Technical Specification change is required.

TSPS 373 Division 3 Power Supply Undervoltage Protection

The design of the Division 3 power supply should have the same two levels of undervoltage protection as that for Divisions 1 and 2 in accordance with SER Section 8.4.4. In the as-built design, only one level of protection (i.e., loss of power (72%)) is provided for Division 3 with a justification. The justification is that starting voltage of the equipment is 75% of rated voltage, which is greater than their setpoint (72%), but accelerating and minimum operating voltages are not provided for NRC staff review. The staff understands that the accelerating and minimum operating voltages should be greater than starting voltage. By letters dated July 3 and August 2, 1984, the licensee committed to implement the second level undervoltage protection for Division 3 power supply at the first refueling outage. The staff concludes that only one level of undervoltage protection would not decrease the availability and reliability of power supplies to Division 3 for this short period of time. The full-power license amendment will require the incorporation of this design feature before startup after the first refueling outage.

The present Technical Specification is in accordance with the as-built design and, therefore, no change is required.

TSPS 808 Diesel Generator Protective Trip Design

The design feature of the diesel generator approved in the SER Section 8.3.1 is that all diesel generator protective trips be bypassed except for diesel engine overspeed and generator differential current. Any other trips retained must use coincident logic to avoid spurious trips. Divisions 1 and 2 diesel generators incorporate a trip on generator ground overcurrent without coincident logic. The lo-lube oil trip is not bypassed on a LOCA signal, but it does use coincident logic. By letters dated July 3 and August 2, 1984, the licensee committed to implement diesel generator protective trips in accordance with that approved in the SER at the first refueling outage. The staff concludes that present design features would not affect the availability and reliability of the diesel generators for this short period of plant operation. The full-power license amendment will require the incorporation of this design feature before startup after the first refueling outage.

The present Technical Specification is in accordance with the as-built design and, therefore, no change is required.

(2) Potential Technical Specification Changes that Depend on Results of Tests, Analyses, or Design Changes

TSPS 081 Reactor Mode Switch

This problem sheet identified potential new requirements in the Limiting Conditions for Operation and Surveillance Requirements for the safety-related functions accomplished by the mode switch. In discussions with the staff, the licensee confirmed that administrative procedures address mode switch surveillance testing and that appropriate remedial actions will be taken if the mode switch were to malfunction. From its review of the existing administrative controls, the licensee has determined that no Technical Specification changes are required. Although Technical Specifications on the mode switch would provide an explicit enforceable set of requirements, the staff recognizes that these would be new requirements. Based on consideration of the licensee's administrative controls, the staff finds that no Technical Specification changes are required at this time.

TSPS 084 Reactor Core Isolation Cooling System Turbine Trips

This problem sheet identified a potential new requirements deficiency in the Limiting Conditions for Operation and Surveillance Requirements for the reactor core isolation cooling system (RCIC). The FSAR identified protective RCIC turbine trips that would prevent RCIC system operation if they were to malfunction or be inoperable. These turbine trips are not included in the Technical Specifications. In discussions with the staff, the licensee confirmed that administrative procedures address RCIC turbine trip surveillance testing and that appropriate remedial action will be initiated if a protective trip malfunctions. Although Technical Specifications on the protective turbine trips would provide an explicit enforceable set of requirements, the staff recognizes that these would be new (generic) requirements. Based on consideration of the licensee's administrative controls, the staff finds that no Technical Specification changes are required at this time.

TSPS 098 Control Room Ventilation System Chlorine Detectors

This problem sheet was initiated to identify a potential Technical Specification change to delete chlorine detectors based on a Bechtel analysis conclusion that control room isolation by the chlorine detectors is not required.

The proposed future Technical Specifications change, pertaining to the operability requirements of the chlorine detector, does not affect the full-power license amendment action. If and when a Technical Specifications change is requested and on receipt of the Bechtel chlorine analysis, the staff will evaluate the proposed change.

TSPS 100 Engineered Safety Features (ESF) Switchgear and Battery Room Temperature Limit

This problem sheet identified a discrepancy between Technical Specifications and design specifications. The maximum temperature in the ESF switchgear and battery rooms should be reduced from 104° to 90°F to provide assurance of a 40-year lifetime of a 750 kVA transformer in the switchgear room.

The effect of continuously operating the transformer at the higher temperature at rated capacity is to reduce the lifetime to 26 years. However, the anticipated infrequent operation at the higher temperature is not expected to have any significant effect on the lifetime of the transformer. Also, because the licensee will incorporate the results of additional analysis into maintenance and surveillance programs to ensure qualification of the transformer and will submit a Technical Specification change to the room temperature limit, if necessary, the staff concludes that no Technical Specification change is necessary at this time.

TSPS 143 Division 1 Battery Capacity

This problem sheet identified the need for Technical Specification changes if additional dc loads are added to the Division 1 dc power supply. The battery capacity is sufficient for present loads but it has a limited reserve capacity. Until the design is changed, no change to the Technical Specification is required.

TSPS 161 Shutdown Cooling Bases

This problem sheet discussed a generic boiling water reactor problem with coolant mixing during cold shutdown (Operational Condition 4) and refueling (Operational Condition 5). Specifically, if shutdown cooling flow is throttled to less than rated to control the cooldown rate, coolant mixing may be inadequate for temperature indication. Operation within the current specification is not hazardous, however, because two subsystems of the ECCS are required to be operable in Operational Condition 4, and also in Operational Condition 5 unless the reactor vessel head is removed and the vessel cavity flooded. In a cold shutdown or refueling condition, any one ECCS subsystem is capable of providing adequate core cooling. When the reactor vessel cavity is flooded, several hours are required before boiling could reduce the reactor vessel water level to the top of the active fuel. No change to the current specification is therefore required for a full-power license amendment.

TSPS 189 Turbine Stop and Control Valve Trip Setpoint

This problem sheet identified a potential change in Technical Specifications regarding the turbine stop and control valve trip setpoints. Notes in the Technical Specifications stated that the final setpoints will be determined during the startup test program. Based on additional review of the Technical

Specifications and the status of the test program, the licensee has determined that no Technical Specification changes are required at this time. Based on the review of the problem sheet, discussions with the licensee, and a review of the Technical Specifications, the staff finds that no Technical Specification changes are required at this time; however, changes may be necessary following the completion of the licensee's startup test program.

TSPS 200 ATWS Recirculation Pump Trip

This problem sheet identified a potential design change regarding the anticipated transients without scram (ATWS) recirculation pump trip instrumentation. The licensee is considering a design change that will provide manual initiation/trip capability. This design feature is not addressed in the Grand Gulf Technical Specifications. From additional review, the licensee has determined that Technical Specification changes are not required until the design modification is completed. Based on the review of the problem sheet and discussions with the licensee, the staff finds that no Technical Specification changes are required at this time.

TSPS 202 Suppression Pool Water Level Monitors

This problem sheet identified a potential change in Technical Specifications regarding the required number of channels for monitoring suppression pool water level. From additional review of the system design and the guidance contained in Regulatory Guide (RG) 1.97, the licensee has determined that the Technical Specifications currently require the appropriate number of operable channels. Based on the review of the problem sheet, discussions with the licensee, and a review of the Technical Specifictions, the staff finds that no Technical Specification changes are required at this time. As discussed in Supplement 4 to the SER, the staff's review of the design's conformance to RG 1.97 is ongoing. Following completion of this review, Technical Specification changes may be required.

TSPS 204 Instrument Setpoints

This problem sheet identified a potential change in Technical Specifications regarding initial and final instrument setpoints. Notes in the specifications state that the final setpoints will be determined during the startup test program. Based on additional review of the Technical Specifications and the status of the test program, the licensee has determined that no Technical Specification changes are required at this time. Based on the review of the problem sheet, discussions with the licensee and a review of the Technical Specifications, the staff finds that no Technical Specification changes are required at this time; however, changes may be necessary following the completion of the licensee's startup test program.

TSPS 207 Radiation Monitor Setpoints

This problem sheet identified a potential change in Technical Specifications regarding initial and final setpoints for radiation monitors. Notes in the specifications state that final setpoints will be determined during the startup test program. Based on additional review of the Technical Specifications and the status of the test program, the licensee has determined that no Technical Specification changes are required at this time. Based on the review of the problem sheet, discussions with the licensee and a review of the Technical Specifications, the staff finds that no Technical Specification changes are required at this time; however, changes may be necessary following the completion of the licensee's startup test program.

TSPS 210 Scram Discharge Volume Level Instrument

This problem sheet identified a potential change in Technical Specifications regarding scram discharge volume level instrumentation. The licensee is implementing a design change that will include installation of scram volume level trip switches. These design features are not addressed in the Technical Specifications. Based on additional review, the licensee has determined that Technical Specification changes are not required until the design modification is completed. Based on its review of the problem sheet and discussions with the licensee, the staff finds that no Technical Specification changes are required at this time.

TSPS 216 Postaccident Monitoring Instrumentation Operability

This problem sheet was initiated to consider potential Technical Specification changes to address the operability of postaccident monitoring instrumentation recommended in NUREG-0737, Supplement 1. It has been determined that all Category I postaccident instrumentation identified in Grand Gulf submittals on RG 1.97, Revision 2, is included in the current Technical Specifications. A final report on the licensee position on RG 1.97, Revision 2, is scheduled for February 1985 in accordance with current regulatory requirements on this item. No change to the Technical Specifications is necessary at this time to address this issue.

TSPS 217 Instrument Surveillance Intervals

This problem sheet identified a potential deficiency in the testing intervals specified in the Technical Specifications. During discussions with the staff, the licensee confirmed that it is participating in the BWR Owner's Group Technical Specification Improvement Program. The purpose of this program is to develop an analytical basis for allowable equipment maintenance down times and intervals between surveillance tests using reliability data, fault tree, and event tree models. Based on its review of the problem sheet and discussions with the licensee, the staff finds no Technical Specification changes required at this time. Adjustments to the Technical Specification surveillance intervals may be justified or required based on the results of this program.

TSPS 218 Instrument Setpoint Methodology

This problem sheet identified a potential deficiency in the methodology used to establih protection systems instrument setpoints. During discussions with the staff, the licensee confirmed that it is participating in a BWR Owner's Group effort to provide more detailed information on their setpoint methodology. This final acceptability of the Grand Gulf setpoint methodology, trip setpoints, and allowable values will be addressed following completion of this effort. In the interim, the staff finds that no Technical Specification changes are required.

TSPS 220 Emergency Lighting and Portable Fire Extinguishers

This problem sheet discussed potential Technical Specifications for emergency lighting and portable fire extinguishers. The NRC staff is currently considering a revision to Standard Technical Specifications for additional Technical Specifications, providing operability and surveillance requirements for portable fire extinguishers and emergency lighting. These additional Technical Specifications have not yet been specifically applied to any plant. Grand Gulf Technical Specifications are consistent with current NRC requirements regarding fire protection. Therefore, no change to Technical Specifications is required at this time.

TSPS 268 Recirculation Pump Trip Response Time

This problem sheet identified a potential change in a Technical Specification regarding the acceptance criterion for recirculation pump trip system response time testing. A footnote to this specification specifies an interim breaker arc suppression time acceptance criterion to be applicable until before startup following the first refueling outage. Based on additional review of the Technical Specification requirements, the licensee has determined that no Technical Specification change is required until before startup following the first refueling outage. Based on the review of the problem sheet, discussions with the licensee, and a review of the Technical Specifications, the staff finds that no Technical Specification changes are required at this time.

TSPS 316 High Drywell Pressure Setpoint

This problem sheet identified a potential deficiency in Technical Specification Table 3.3.2-2 regarding the high drywell pressure ECCS setpoint. Recent design specifications from GE (the NSSS vendor) provided a setpoint that is more conservative than that now in Technical Specifications; however, calculations supporting the new value have not been provided by GE. From a review of the information provided by GE, the licensee has determined that no Technical Specification changes are required at this time. During discussions with the staff, the licensee confirmed that it is participating in a BWR Owner's Group effort to provide more detailed information on their setpoints and setpoint methodology. The final acceptability of the Grand Gulf setpoints methodology, trip setpoints, and allowable values including the setpoint for the high drywell pressure will be addressed following completion of this effort. In the interim, the staff finds that no Technical Specification changes are required for this instrument.

TSPS 326 Fuel Pool Cooling Pump Room Backup Coolers

License Condition 2.C.(29) requires plant shutdown if spent fuel is placed in the spent fuel pool before installation and operability of the backup room coolers for the fuel pool cooling pump room. This problem sheet was initiated to consider adding a Technical Specification instead of the license condition. The staff concludes that the license condition provides adequate assurance with respect to the single failure criteria and cooling the pump room and thus no Technical Specification need be provided at this time. When the installation of the backup room coolers is completed, the Technical Specifications must be changed.

TSPS 347 Control Rod Bypass Switches

This problem sheet identified potential new requirements regarding the use of the individual control rod bypass switches. From its review of the Technical Specifications and plant administrative controls, the licensee has determined that the proposed Technical Specification would, for the most part, consolidate existing requirements. In discussions with the staff, the licensee confirmed that administrative procedures govern bypass switch operation. Although Technical Specifications on the bypass switch would provide an explicit enforceable set of requirements, the staff recognizes that these would be new (generic) requirements. Based on consideration of the licensee's administrative controls, the staff finds that no Technical Specification changes are required at this time.

TSPS 357 Recirculation Pump Trip Bypass

This problem sheet identified potential new requirements in Technical Specifications regarding the automatic bypass circuitry for the end-of-cycle recirculation pump trip (EOC-RPT) system. There are no Technical Specification requirements to calibrate or functionally test the bypass circuits or Technical Specification requirements that prescribed remedial action in the event of inoperability. From a review of the system design and plant administrative controls, the licensee has determined that no changes are required. In discussion with the staff, the licensee confirmed that periodic at-power testing of the EOC-RPT and reactor trip system circuits would detect failures that could lead to inadvertent bypass conditions, and that plant administrative controls provide for periodic surveillance of the bypass circuits and actions to be taken if the bypass circuitry is inoperable. Although Technical Specifications on the bypass circuitry would provide an explicit enforceable set of requirements, the staff recognizes that these would be new (generic) requirements. Based on consideration of the licensee's administrative controls, the staff finds that no Technical Specification changes are required at this time.

TSPS 362 Reactor Core Isolation Cooling System Time Delay Relays

This problem sheet identified a potential change in Technical Specifications, regarding time delay relays currently being added to the reactor core isolation cooling (RCIC) system initiation instrumentation. These time delay relays are not specifically addressed in the Technical Specifications. From additional review, the licensee has determined that Technical Specification changes are not required until the design modification, including testing, is completed. Based on its review of the problem sheet and discussions with the licensee, the staff finds that no Technical Specification changes are required at this time.

TSPS 366 Containment Spray Response Time Surveillance

This problem sheet identified potential new requirements regarding the containment spray system response time surveillance requirements. Presently, plant administrative controls govern the surveillance performed to ensure that the combination of instrument response times and opening times for the spray valves do not exceed the time limit derived from the FSAR accident analysis. Although Technical Specifications on response time testing for the containment spray initiation instrumentation would provide an explicit enforceable set of requirements, the staff recognizes that these would be new (generic) requirements.

Based on consideration of the licensee's administrative controls, the staff finds that no Technical Specification changes are required at this time.

TSPS 370 Reactor Protection System Bypass Instrumentation

This problem sheet identified potential additional Technical Specifications regarding reactor protection system bypass instrumentation. The Technical Specifications do not include Limiting Conditions for Operation or Surveillance Requirements on these bypass circuits. Plant administrative controls in conjunction with Technical Specifications that address the reactor protection system instrumentation provided a set of requirements applicable to the bypass circuits. In discussions with the staff, the licensee outlined a long-term program for resolution of this issue that includes a BWR Owner's Group evaluation. In the interim, the licensee has determined that no Technical Specification changes are required. Although Technical Specifications that specifically address the reactor protection system bypass circuitry would provide an explicit enforceable set of requirements, the staff recognizes that these would be new requirements that must be approved and incorporated into the BWR Standard Technical Specifications (NUREG-0123, Rev. 3) before implementation in specific plants. Based on consideration of the licensee's administrative controls, the current Technical Specification requirements and the licensee's commitment to the long-term BWR Owner's Group Technical Specification improvement effort, the staff finds that there are no Technical Specification changes required at this time.

TSPS 374 Postaccident Monitoring Instrumentation Completeness

This problem sheet identified a potential deficiency in a Technical Specification regarding the completeness of the list of postaccident monitoring instrumentation. From additional review of system design and Technical Specifications, the licensee has determined that each instrument classified as Category I by the licensee (as defined in Regulatory Guide 1.97) is included in the current specification. Based on the review of the problem sheet, discussions with the licensee and a review of the Technical Specifications, the staff finds that no Technical Specification changes are required at this time. As discussed in Supplement 4 to the SER, the staff's review of the design's conformance to RG 1.97 is ongoing. Following completion of this review, Technical Specification changes may be required.

TSPS 377 Control Room Air Inleakage

This problem sheet addressed tests conducted by the licensee that indicate control room in-leakage in excess of the design value. Amendment 11 to the Grand Gulf operating license added a license condition directing that additional information be provided to support an increased control room leak rate to address this concern. This problem sheet concerned the need to add control room inleakage requirements to the Technical Specifications. The staff has evaluated this concern and determined that the license condition adequately addresses the resolution of this issue. It has been determined that no changes to the Technical Specification are required concerning this item at this time, although final resolution of the license condition may result in future Technical Specification changes.

16.3.2 Technical Specification Changes That Would Reduce Conservatism

The problem sheets that are evaluated in this section identified changes that were found by the staff to reduce the conservatism in the Technical Specifications without adequate justification.

TSPS 008 Single Recirculation Loop Operation

This problem sheet stated that Technical Specification changes that would be needed to permit operation with only one recirculation loop functioning will be requested in the future. Single-loop operation is an operational improvement only and is not needed for safe plant operation. Operation within the current specifications is conservative because the current specifications require reactor shutdown when one or more recirculation loops are inoperable. No Technical Specification changes are therefore required for safe operation.

TSPS 029 Source Range Monitor Surveillance Tests

This problem sheet identified discrepancies between the surveillance test procedures and a Technical Specification regarding surveillance requirements for testing the source range monitors. Technical Specifications require a channel functional test of the source range monitors before moving the reactor mode switch out of the Shutdown position. The surveillance procedures require that the reactor mode switch be placed in the Refuel position to perform the surveillance test. From a review of the system design and discussions with GE (the NSSS vendor), the licensee has found alternate methods of performing the surveillance test that do not require mode switch movement. Surveillance procedure revisions are currently being prepared by the licensee. Based on its review of the problem sheet and discussions with the licensee, the staff finds that no Technical Specification changes are required.

TSPS 030 ECCS Instrumentation

This problem sheet identified differences between the Draft BWR/6 Standard Technical Specifications and the Action Statements and Surveillance Requirements of Technical Specifications regarding the high-pressure core spray (HPCS) line break detection instrumentation, emergency core cooling system (ECCS) discharge line "keep filled" pressure alarm instrumentation, and ECCS header "delta pressure" instrumentation. The Draft BWR/6 Technical Specifications have not been approved by the staff and therefore are not staff guidelines. The plant-specific instrumentation described herein is required by the staff's safety analysis. Deletion of the above-mentioned specifications was considered by the licensee as an operational improvement to the Technical Specifications. Based on its review of the problem sheet and discussions with the licensee, the staff finds that no Technical Specification changes are required for safe operation.

TSPS 042 Recirculation Flow Nomenclature

This problem sheet identified an alternate method of measuring recirculation loop flow mismatch that may provide operational enhancement. The current Technical Specifications accomplish the same objective, and the proposed changes provide no apparent safety improvement. The licensee has decided not to request this change. No change to the specification is therefore required.

TSPS 046 RHR Shutdown Cooling During Refueling

This problem sheet discussed a potential Technical Specification change to allow refueling without an RHR shutdown cooling loop in operation for an extended (to be determined) period of time. The present specification requires that a shutdown cooling loop be in operation during refueling.

The licensee believes there may be a problem resulting from an RHR loop operating during refueling because of a possible reduction in water clarity and visibility. Changing the specification to allow refueling without an RHR loop operating may facilitate refueling operations somewhat, but leads to no apparent safety enhancement. The present specification is considered more conservative in that decay heat removal is continuously required. No change to the specification is, therefore, required for safe operation.

TSPS 048 Average Power Range Monitor (APRM) and Rod Block Setpoints

This problem sheet stated that analyses are being performed by GE to extend the operating domain by changing the setpoints of the APRM and the rod blocks. Licensee plans to request changes to Technical Specifications when the analyses are completed. This is strictly an operational improvement and results in no apparent safety improvement. The current specification is more conservative than would be used in the extended domain. No change is therefore required.

TSPS 104 Control Room Command Function

This problem sheet identified a concern raised by the licensee involving the individual responsible for the control room command function. As specified in the current Technical Specifications, the control room command function is the responsibility of the Shift Superintendent. However, the licensee felt that it would be more appropriate to assign this function to the Shift Supervisor, thus allowing the Shift Superintendent more flexibility in his duties (i.e., by no longer requiring him to designate an individual to assume his duties each time he leaves the control room).

The staff has reviewed the licensee's FSAR description of plant personnel responsibilities and authorities and concludes that the Technical Specification as written conforms to the requirements of NUREG-0737 concerning control room command; therefore, no Technical Specification change is warranted.

TSPS 141 Diesel Generator Surveillance Test Frequency

This problem sheet identified a potential operational improvement regarding periodic surveillance testing of the two 7000-kW Transamerica Delaval, Inc. (TDI) diesel generators and the one 3300-kW EMD diesel generator. The Technical Specifications for surveillance tests, in accordance with Section C.2.d of RG 1.108 ("Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Plants," Rev. 1, August 1977) require more frequent tests for all diesel generators at the plant if failures during periodic testing exceed a certain value. The licensee raises the question of whether different types and sizes of diesel generators should be treated separately. This operational improvement has not been reviewed and approved for the BWR Standard Technical Specifications and is not required for safe operation.

TSPS 142 Diesel Generator Fast Start Testing

This problem sheet identified a potential operational enhancement regarding the Technical Specification requirements for periodic testing of diesel generator fast cold starting. Fast cold starts demonstrate capability to meet design-basis-accident requirements. The licensee believes that deletion of the requirement for fast cold starts, or decreasing the required frequency, will extend the diesel generator engine lifetime. Surveillance testing requirements in the Technical Specifications are in compliance with regulatory requirements and the BWR Standard Technical Specifications. This operational improvement, which has not been justified, is not required for safe operation. Therefore, no change to Technical Specifications is required at this time.

TSPS 159 Reactor Coolant Leakage in Drywell

This problem sheet identified a potential operational improvement regarding elimination of the particulate monitor in the drywell, which is used for detecting reactor coolant leakage. The licensee notes that other licensees of BWR plants have removed the atmosphere particulate monitoring system from their Technical Specifications and intends to perform an evaluation to demonstrate that the particulate monitoring system is not needed at Grand Gulf. No Technical Specification change has been proposed by the licensee. Use of the airborne radioactivity particulate monitoring system to monitor leakage is conservative. Therefore, no Technical Specification change is required.

TSPS 181 Channel Functional Test for Reactor Protection System Electric Power Monitoring Assemblies

By letter dated June 21, 1984, the licensee proposed a change to Surveillance Requirement 4.8.4.3.a. to require a channel functional test of the reactor protection system (RPS) electric power monitoring assemblies (EPA) only when the plant is in Cold Shutdown for a period of more than 24 hours, unless performed in the previous 6 months. The licensee provided a justification that this change will lessen the potential of an accidental reactor trip and isolation, as a result of switching to the alternate power supply for testing.

The testing of the EPA units actually results in a 1/2 scram condition and a 1/2 isolation signal, which also occurs many times monthly for RPS and isolation channel tests. The additional actions that occur for the EPA unit testing are the isolation of shutdown cooling and head spray modes of RHR system (manually actuated system) and the momentary loss of some radiation monitoring channels. The staff does not see any overriding safety reason for this Technical Specification change. The justification provided by letters of June 21, 1984, July 13, 1984, and July 25, 1984 is considered inadequate, and therefore, the change request is denied.

TSPS 214 Control Rod Drive Scram Accumulator Surveillance

This problem sheet concerned a Technical Specification Surveillance Requirement for control rod drive scram accumulator check valves. The Technical Specification requires that acceptable leakage past each accumulator check valve be determined by measuring the time, for up to 10 minutes, that each check valve maintains accumulator pressure above the alarm setpoint. By letter dated June 22, 1984, the licensee requested that the Surveillance Requirement be deleted because the leakage test acceptance criteria were not appropriate. GE (the NSSS vendor) indicated that the present Technical Specification satisfies the design intent. The NRC staff is currently considering a potential change to this Technical Specification. However, until a revision is approved for BWR plants, the current Technical Specification will be used. Therefore, no Technical Specification change will be made at this time.

TSPS 221 Reactor Coolant System Leakage

This problem sheet identified a difference between the BWR Standard Technical Specification and the Technical Specifications regarding reactor coolant system leakage limits. The Technical Specification allows a total leakage rate of 30 gpm that was accepted in the SER. The BWR Standard Technical Specification is less restrictive. The licensee did not provide adequate justification to reduce the conservatism in this specification. Therefore, the staff concludes that no Technical Specification modification is required.

TSPS 289 Supervision of Core Alterations

By letter dated June 22, 1984, the licensee proposed a change in Section 6.2.2 of the Technical Specifications which would permit the maintenance foreman to observe and directly supervise the replacement, under the reactor vessel, of neutron monitors rather than requiring a senior reactor operator (SRO) to perform this task as presently required by the Technical Specifications. The SRO is trained and has passed examinations on core alterations, which include movement of in-core instrumentation, whereas the maintenance foreman does not typically receive such training and examination. Therefore, the proposed change is unacceptable. The present Technical Specifications are acceptable.

TSPS 321 Suppression Pool Limiting Temperature

This problem sheet concerned a difference between Technical Specifications and the BWR/6 Standard Technical Specification (STS). This Action Statement in the BWR/6 STS would allow pool temperature to exceed 95°F when reactor thermal power is less than 1% of rated. The NRC staff has not approved the draft BWR/6 STS. Because the Grand Gulf Technical Specification is more conservative than the draft BWR/6 STS, no change to the Grand Gulf Technical Specification is required.

TSPS 330 Accident Monitoring Instrumentation Channel Check Frequencies

This problem sheet described a potential operational enhancement to decrease frequency of channel checks of accident monitoring instrumentation, based on the draft BWR/6 STS requirements. The NRC staff has not approved the draft BWR/6 STS.

From additional review of the Technical Specification requirements and review of the recommendations contained in the BWR/6 Standard Technical Specifications, the licensee has determined that no Technical Specification changes are required. Based on the review of the problem sheet, discussions with the licensee, and a review of the Technical Specifications, the staff finds that no Technical Specification changes are required.

TSPS 332 Condensate Storage Tank Water Level

This problem sheet identified a discrepancy between the volume of water in the condensate storage tank and level indication on the tank. The difference (1 ft of elevation) is the difference between the instrument zero level elevation and the bottom of the tank. The Technical Specification now uses the instrument elevation, similar to other water level specifications for the plant. The instrument elevation indicates less water than would actually be in the tank. Thus, the instrument level is conservative. A change in the Technical Specifications to reference the bottom of the tank could result in confusion and potential unsafe plant operation. Therefore, no change to the Technical Specification is required.

TSPS 380 Low Power Physics Tests

This problem sheet identified a potential operational enhancement regarding the Technical Specifications that address special test exceptions on vessel, containment, and drywell integrity, and entry into the startup operational condition for low power physics testing. The specification as written is the standard one for all types of BWRs and only permits low power physics testing as part of the open vessel testing. The staff concludes that no change is required in the Technical Specification.

TSPS 805 Sodium Pentaborate Volume

This problem sheet identified a discrepancy between the Technical Specifications and the FSAR regarding the volume of sodium pentaborate. The FSAR Figure 9.3-26 specifies that the standby liquid control system (SLCS) will be able to deliver 4,170 gallons of sodium pentaborate solution. The Technical Specifications identify a minimum volume of 4,587 gallons of sodium pentaborate. The FSAR implies that operation of the redundant SLCS pump will be demonstrated when an SLCS pump is out of service for maintenance. The Technical Specification of the volume is greater than the FSAR-specified delivered volume, thus the Technical Specification is conservative. In addition, the tank volume and pump delivery are not contradictory. The SLCS pumps will be demonstrated operable every 31 days, per normal surveillance requirements. No additional surveillance is necessary. The staff concludes that no changes to the FSAR or the Technical Specifications are required.

16.3.3 Final Safety Analysis Report (FSAR) Changes

(1) FSAR Changes Made in Amendment 58 to the FSAR

TSPS 151 and 800 Average Power Range Monitor Trip Allowable Value

This problem sheet identified a discrepancy between the Technical Specification's allowable value for the average power range monitor (APRM) neutron flux high trip (120%) and the value approved by the staff's SER (112.5%). From additional review of the FSAR, Technical Specifications, and instrument specifications provided by GE (the NSSS vendor), the licensee has determined that the FSAR should be changed but not the Technical Specifications. FSAR Amendment 58 corrected this error. Based on its review of the problem sheet and discussions with the licensee, the staff finds that no Technical Specification changes are required.

TSPS 152 Shutdown Margin Requirement

This problem sheet related to discrepancies between the shutdown margin required in the Technical Specifications and the shutdown margin required in the FSAR. The Technical Specifications allow two different methods of determining the highest worth rod for purposes of measuring the shutdown margin. If the strongest rod is determined analytically, the required shutdown margin at the time in cycle when it is a minimum must be $\geq$ 0.38% reactivity change. If the strongest rod is determined by test, the required shutdown margin is only 0.28% reactivity change. This is not in agreement with the FSAR that required only 0.25% shutdown margin and did not differentiate between the two ways of determining the strongest rod.

The Technical Specification as written is the standard one for boiling water reactors and is conservative with respect to the FSAR. The staff concludes, therefore, that no change is required in the Technical Specification and its bases. Grand Gulf has submitted Amendment 58 to the FSAR, which makes it consistent with the Technical Specifications.

TSPS 234 Suppression Pool Level Instrumentation

This problem sheet identified a potential discrepancy between the FSAR descriptions and the Technical Specifications that address the suppression pool level instrumentation. The Technical Specifications regarding accident monitoring instruments, emergency core cooling systems, and containment systems address the surveillance requirements applicable to the suppression pool level instrumentation. However, the Technical Specifications do not identify each instrument by the unique plant-specific identification number to avoid confusion regarding which of the eight instrument channels are addressed by a particular Technical Specification. From additional review of the design, the Technical Specifications, and the plant administrative procedures, the licensee has determined that the procedures implementing the Technical Specification's requirements correctly identify each instrument.

To allow flexibility in operation and to accommodate design modifications, the NRC staff has avoided specifying instruments by unique plant-specific identification number in the Technical Specifications. In FSAR Amendment 58, changes were made to clarify the design description. Accordingly, based on the review of the problem sheet, review of the design and discussions with the licensee, the staff finds that no Technical Specification changes are required.

TSPS 252 Containment and Drywell Air Volume

This problem sheet identified discrepancies between the Technical Specifications and the FSAR regarding containment and drywell net free air volume. The licensee has stated that the Technical Specification values are consistent with plant design. Also, the FSAR analyses were performed using the Technical Specification values. The magnitudes of the discrepancies are numerically insignificant because use of either set of values would not affect the analytical results significantly. However, the FSAR was revised in Amendment 58 to reflect the correct values. The staff finds that a Technical Specification change is not required.

TSPS 260 Containment Air Temperature

This problem sheet identified an inconsistency between the containment air temperature limit in Technical Secifications compared to that in the FSAR. The Technical Specifications indicate that the containment average air temperature shall not exceed 90°F. In the FSAR, the drywell and containment pressure evaluations assumed an initial containment air temperature of 80°F. However, additional analyses associated with Mark III containment issues (Humphrey concerns) have confirmed that the containment atmosphere design temperature during LOCA conditions is not exceeded when the initial containment atmosphere temperature is equal to 95°F. Results of these additional analyses were included in FSAR Amendment 58. Based on these evaluations and the conservative nature of the FSAR analyses, the NRC staff concludes that the Technical Specification limit on containment average air temperature of 90°F would not compromise containment integrity under accident conditions. Therefore, no change to the Technical Specifications is required.

TSPS 291 Main Steam Isolation Valve Closure Time

This problem sheet identified an apparent discrepancy between the 5.5-second time for main steam isolation valve (MSIV) closure stated in the FSAR and the instrument response time of 1 second allowed by the Technical Specification for MSIV isolation signals reactor vessel water level 1, main steam line radiation high and main steam line pressure low. By assuming an instrument response time of 1 second, plus an MSIV closure time of 5 seconds, as allowed by the Technical Specification, a total MSIV closure time of 6 seconds could occur, as opposed to the 5.5-second closure time stated in the FSAR. It has been determined that the values of the Technical Specification are adequate and that they represent no safety concerns. The 5.5-second MSIV closure time stated in the FSAR is assumed only for a main steam line guillotine break outside containment. The primary MSIV closure signal used to initiate MSIV closure for this accident would be the main steam line high flow signal, which has a required response time of 0.5 seconds and is therefore consistent with the FSAR. Although other MSIV closure signals could occur in the main steam line break scenario, they are required to perform no mitigating function. FSAR Amendment 58 clarified that the 5.5-second detection and closure time applies only for a main steam line break. The staff finds this FSAR change to be an acceptable manner in which to clarify that there is no real discrepancy between the FSAR and the Technical Specifications. Therefore, no Technical Specification change is required.

TSPS 300 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Limits

This problem sheet identified a discrepancy between MAPLHGR limits given in the Technical Specifications and those provided in the FSAR. The licensee has indicated in Amendment 58 to the FSAR that the value in the FSAR is a typographical error. The FSAR has been amended to correct this error. The Technical Specification value is correct, and no change to the Technical Specifications is required. The FSAR amendment is acceptable because correction of the typographical error does not affect the results of the associated safety analyses.

TSPS 339 Operations Personnel Titles

An inconsistency existed between operations personnel position titles in the FSAR and the Technical Specifications. The FSAR used the title "Non-Licensed Operator" for the positions identified in Technical Specifications as "Auxiliary Operator" or "Nuclear Operator-B". Amendment 58 to the FSAR, submitted on May 18, 1984, defines Non-Licensed Operator as Auxiliary Operator or Nuclear Operator-B, thereby resolving the inconsistency. Based on this resolution, no Technical Specification change is necessary.

TSPS 340 Operations Personnel Break Time

The minimum break time for unit operations personnel who perform safety-related functions was specified as 8 hours in GGNS Technical Specifications and as 12 hours in the FSAR. The 8-hour minimum break time is acceptable to the staff, and the FSAR has been revised, by Amendment 58, to be consistent with the Technical Specifications. Therefore, no Technical Specification change is necessary.

TSPS 801 Containment to Auxiliary Building Differential Pressure

This problem sheet identified a discrepancy between the FSAR and Technical Specifications regarding limits on containment to auxiliary building differential pressure. The licensee has found that the Technical Specification limits on containment to auxiliary building differential pressure are the correct values. In Amendment 58, the FSAR has been changed appropriately to correct the normal operating containment to auxiliary building differential pressure. The normal operating conditions are presented in the FSAR for information only. Because the safety analyses results or conclusions are not affected by this change, the staff concludes that the FSAR change is acceptable. The staff finds that a Technical Specification change is not required.

TSPS 802 Recirculation Pump Trip Setpoint

This problem sheet identified a discrepancy between the Technical Specifications and the FSAR with regard to the power level at which the end-of-cycle recirculation pump trip must be operable. The licensee has determined that this question was addressed in Amendment 55 to the FSAR in Section 15.2. A corresponding revision to FSAR Section 7.6.1.8.1 was not made in Amendment 55, however. The Technical Specification value is correct and consistent with FSAR Section 15.2, and no Technical Specification change is, therefore, required. FSAR Section 7.5.1.8.1 was updated in Amendment 58.

TSPS 806 Minimum Critical Power Ratio Limit

This problem sheet concerned a discrepancy between the Technical Specifications and the FSAR with respect to the operating limit minimum critical power ratio (MCPR) required for the fuel misloading event. The FSAR value of 0.13 for the change in CPR caused by this event implies an operating limit MCPR value of 1.19 compared to the Technical Specification value of 1.18. In Amendment 58 to the FSAR, the licensee provided an updated analysis of the fuel misloading event. The resultant change in CPR is 0.10. The staff has reviewed this analysis and found it to be acceptable. This finding is based on the fact that acceptable methods have been used to perform the analysis, appropriate assumptions are made with respect to possible misloading events, and acceptable results are obtained. The staff therefore concludes that no change in the Technical Specification value of the operating limit MCPR is required.

TSPS 812 Main Steam Tunnel Temperature Timer

This problem sheet identified a deficiency in the FSAR in that no description of the main steam tunnel temperature timer is included in the FSAR, although it is included in the Technical Specifications. The FSAR has subsequently been revised in Amendment 58 to clarify the inclusion of a time delay before RCIC isolation on high main steam tunnel temperature. The staff has reviewed the safety analyses and determined that they are unaffected by this FSAR change. The Technical Specifications therefore require no change.

TSPS 816 Main Steam Line Flow Setpoint

This problem sheet identified a discrepancy between the Technical Specifications and the FSAR regarding the set point for the main steam line-high flow instrumentation. The FSAR stated that the maximum allowable setpoint was 133.5 psid. The Technical Specifications use 169 psid, which is the value used in FSAR analyses. From its review of the FSAR, Technical Specifications and instrument specifications provided by GE (the NSSS vendor), the licensee has determined that no Technical Specification changes are required. Amendment 58 corrects the error in the FSAR. Based on its review of the problem sheet and discussions with the licensee, the staff finds that no Technical Specification changes are required.

TSPS 817 Standby Gas Treatment System Capacity

This problem sheet identified a deficiency in the FSAR in that the FSAR did not describe the capability of the standby gas treatment system (SGTS) to maintain negative pressure, postulating failure of a single 4-in. line penetration or failure of all nonqualified lines 2 in. and smaller. FSAR Amendment 58 incorporates the above information. The purpose of the revision was to clarify the SGTS drawdown capability. Based on a review of the related Technical Specifications, the staff finds that the Technical Specifications are not affected by this FSAR change.

TSPS 819 Seismic Instruments

This problem sheet identified an error in the FSAR in that a response spectrum analyzer was incorrectly labeled as a triaxial response spectrum recorder. Accordingly, a change to the FSAR was submitted by the licensee in FSAR Amendment 58 to correctly identify the response spectrum analyzer. This change did not affect the Technical Specification on seismic monitoring instrumentation, which was verified to be correct. No Technical Specification change is required.

TSPS 820 Reactor Core Isolation Cooling System Setpoints

This problem sheet identified discrepancies between the Technical Specifications and the FSAR regarding the reactor core isolation cooling (RCIC) system instrumentation. The setpoints for six monitored parameters providing input to the RCIC system instrumentation were not in agreement. From additional review of the FSAR, Technical Specifications, and instrument specifications provided by

GE (the NSSS vendor), the licensee has determined that no Technical Specification changes are required. FSAR Amendment 58 deleted the values of setpoints from the FSAR, thus eliminating the possibility of conflicts with setpoints in the Technical Specification. This is acceptable to the staff. Based on its review of the problem sheet and discussions with the licensee, the staff finds that no Technical Specification changes are required.

TSPS 822 Standby Gas Treatment System Performance

This problem sheet concerned several apparent inconsistencies between the FSAR and Technical Specification 3/4.6.6.3 concerning standby gas treatment system (SGTS) flow testing. Some of these inconsistencies involve changes to the FSAR but not to the Technical Specifications, which are correct and provide an acceptable level of safety. In particular, Amendment 58 to the FSAR corrected the time for secondary containment drawdown from 101 to 120 seconds. Other changes to the FSAR revise the licensee commitment from Revision 1 to Revision 2 of Regulatory Guide 1.52, which contains the most current regulatory guidance, and correct the SGTS long-term flow rate from 2300 to the analytical limit of 4500 cfm. The staff concludes that these FSAR changes are acceptable and that no change to the Technical Specifications is required.

TSPS 823 Auxiliary Building Isolation Dampers and Valves

This problem sheet identified a deficiency in the FSAR regarding isolation dampers and valves needed to isolate the auxiliary building following an accident to provide secondary containment. The completeness of the Technical Specification cannot be verified by information in the FSAR because certain ventilation system isolation dampers and a valve in an RHR branch line listed in Technical Specifications are not listed in the FSAR. The licensee has affirmed that the Technical Specification listing is consistent with plant design. In FSAR Amendment 58, the FSAR has been modified accordingly. Therefore, Technical Specification changes are not needed.

TSPS 828 Reactor Core Isolation Cooling System Initiation Instrumentation

This problem sheet identified a discrepancy between the Technical Specifications and the FSAR regarding the reactor core isolation cooling (RCIC) system initiation instrumentation. The FSAR indicated that the RCIC system is initiated on receipt of a reactor vessel low water level signal. The Technical Specifications indicate that the RCIC system is initiated on receipt of a reactor vessel low-low water level signal. From additional review of the design, the licensee has confirmed that the Technical Specifications correctly reflect the RCIC system initiation on low-low level and determined that a change to the FSAR is required. This change was made in FSAR Amendment 58. Based on its review of the problem sheet and discussions with the licensee, the staff finds that no Technical Specification changes are required.

TSPS 830 Reactor Vessel Thermal Cycle Limit

This problem sheet identified a discrepancy between the Technical Specifications and the FSAR regarding a limitation on recirculation pump flow increases under certain reactor conditions. This limitation is imposed to limit thermal cycles on reactor vessel nozzles. FSAR Section 5.3.3.6 stated that if the coolant temperature difference between the dome and the bottom head drain exceeds

145°F, neither reactor power level nor recirculation pump flow shall be increased. This temperature limit value is for BWR/4 and BWR/5 plants and is incorrect for BWR/6 plants. The correct value for BWR/6 plants is 100°F, as specified in the Technical Specifications. Thus, the applicant submitted a revised temperature limit value of 100°F in FSAR Amendment 58 dated May 18, 1984.

The staff has reviewed the proposed temperature limit change in the FSAR Amendment 58 and determined that the proposed revision is a correction of error and does not reduce safety margins in the plant design and will not increase the probability or consequences of either a new or a previously analyzed accident. Therefore, the proposed revision is acceptable. No change to the Technical Specification value is required.

TSPS 831 Reactor Protection System Response Times

This problem sheet identified discrepancies between the Technical Specifications and the FSAR regarding reactor protection system response times for four trip functions: reactor vessel low water level, reactor vessel high water level, turbine stop valve closure, and turbine control valve fast closure. From additional review of the design specifications provided by GE (the NSSS vendor), the licensee has confirmed that the Technical Specification values are correct and determined that a change to the FSAR was required. This change was made in FSAR Amendment 58. Based on the review of the problem sheet and discussions with the licensee, the staff finds that no Technical Specification changes are required.

TSPS 832 Standby Gas Treatment System Design

This problem sheet identified an inconsistency between the Technical Specifications and the FSAR concerning the standby gas treatment system (SGTS). The Technical Specification indicates that the two SGTS trains are divisionally independent, whereas the FSAR stated that any initiation signal would start both trains. The FSAR was incorrect and a change to applicable portions was included in Amendment 58. The staff finds these changes to the FSAR to be acceptable. The current Technical Specifications are correct in this regard; therefore, no Technical Specification change is required.

TSPS 833 Diesel Generator Load Reject Capability

This problem sheet identified discrepancies between the Technical Specification and the FSAR regarding the values of maximum pump loads that could be rejected without exceeding overspeed limits. The Technical Specifications have load values that conform to tested load values and are therefore correct. In Amendment 58 to the FSAR, the licensee changed the FSAR load values to be consistent with the Technical Specifications. Therefore, no Technical Specification change is required.

(2) FSAR Changes To Be Made at the Next Annual Update of the FSAR

TSPS 025 Main Steam Isolation Valve (MSIV) Minimum Closing Time

This problem sheet identified an inconsistency between the FSAR and the Technical Specifications regarding the minimum MSIV closure time requirements. The FSAR requires that the average of valve closure times be greater than

3.0 seconds with the fastest valve closure time greater than 2.5 seconds. The Technical Specifications require that each individual valve have a minimum closure time greater than 3.0 seconds.

The staff has reviewed the differing requirements and concludes that no change is required to the Technical Specifications because the Technical Specification is more conservative and is consistent with FSAR safety analyses. The staff requires that the FSAR be updated to reflect values in the Technical Specifications.

TSPS 259 Reactor Vessel Cyclic or Transient Limits

This problem sheet identified a discrepancy between the reactor vessel cyclic or transient limits in a Technical Specification design feature table, Table 5.7.1-1, in the FSAR. FSAR Table 3.9-1 identifies the transients and number of cycles considered in the design and fatigue analysis of the reactor vessel and internals. The cyclic or transient limits in the Technical Specification table are in agreement with limits in FSAR Table 3.9-1. FSAR Table 5.2-11 indicates the anticipated operating thermal cycles for the reactor coolant pressure boundary. There is a difference in the number of transients identified in the FSAR table and the Technical Specification table. The number of cycles in the Technical Specifications is conservatively lower than that in the FSAR. Therefore, no Technical Specification change is required. However, the staff requires that the FSAR be made consistent with the Technical Specifications in the next annual update of the FSAR.

TSPS 305 Flood Protection

This problem sheet identified the potential need for a new Technical Specification for flood protection. Supplement 2 to the SER required that permanent modifications be made to grade-level mechanical door seals in the control building, diesel generator building, and service water pumphouse and to floor penetrations in the service water pumphouse. These modifications are needed to provide adequate flood protection for safety-related equipment used for safe shutdown of the plant. License Condition 2.C.(5) required temporary protection by sand bags and plans for a permanent solution before exceeding 5% power. The licensee has made the permanent modifications and they have been inspected and found acceptable by the NRC staff. The staff requires the FSAR to be updated to reflect the permanent modifications; however, a new Technical Specification is not required.

TSPS 327 Radiation Monitor Calibration

This problem sheet identified an inconsistency between the Technical Specifications and the FSAR regarding calibration of containment and drywell radiation monitors. The FSAR states that "during each refueling outage, the instrument will be returned to the manufacturer for recalibration" whereas the Technical Specifications state that the channel calibration of the instrument will be performed with an installed or portable gamma source at the plant during each refueling outage. Calibration with a gamma source at the site is acceptable to the staff. The staff requires that the FSAR be revised to reflect Technical Specifications in the next annual FSAR update. Therefore, no Technical Specification change is required.

TSPS 372 Manual Isolation of Primary Containment

This problem sheet identified a discrepancy between the as-built plant and the Technical Specifications regarding the manual initiation of a valve group 6A isolation. According to the Technical Specifications, valve group 6A receives a closure signal from manual initiation of primary containment isolation. Eight valves in group 6A did not close from manual initiation of primary containment isolation during a surveillance test. In discussions with the staff, the licensee has stated that plant modifications have been completed to provide manual initiation capability. Based on its review of the problem sheet and discussions with the licensee, the staff finds that Technical Specification changes are not required. The staff requires the FSAR to be amended to reflect the revised design.

TSPS 376 Standby Gas Treatment System Flow Limit

The licensee initiated this problem sheet in response to an NRC staff comment that a Technical Specification surveillance requirement should be revised to require a maximum standby gas treatment system (SGTS) flow rate not exceeding 2300 cfm during verification of secondary containment integrity. The current specification requires a flow rate not exceeding 4000 cfm while maintaining $\geq$ 0.266 in. of vacuum water gauge in the secondary containment. The limit of 2300 cfm was believed to be correct for a plant of design similar to Grand Gulf. It has been verified that the correct design flow rate of the Grand Gulf SGTS is 4000 cfm based on an analytical limit of 4500 cfm and that the thickness of the charcoal beds in the SGTS is adequate to ensure the 99% iodine removal efficiency required of the adsorbers to maintain offsite dose rates within all applicable limits and regulatory guidelines at an SGTS flow rate of 4000 cfm. Therefore, no change to the Technical Specifications is required. The staff requires the FSAR to be changed to reflect the correct flow rate.

TSPS 807 Radiation Monitor Calibration Frequency

This problem sheet identified inconsistencies between the Technical Specifications and the FSAR regarding calibration frequencies of radiation monitors. In one section, the FSAR requires calibration "...annually during plant operation or during the refueling outage if the detector is not readily accessible". In another section, the FSAR requires calibration "...annually...." Technical Specifications require calibration during a refueling outage. The Technical Specification frequencies are consistent with the frequencies in the Draft BWR/6 Standard Technical Specifications and are acceptable to the staff. The staff requires the FSAR to be updated to reflect the frequencies used in Technical Specifications.

TSPS 810 Circuit Breaker Trip Setpoints

This problem sheet identified an inconsistency between the Technical Specifications and the FSAR regarding circuit breakers. Trip setpoints for 6.9-kV circuit breakers contained in the FSAR are inconsistent with the information contained in the Technical Specifications. The staff finds the values in the Technical Specifications to be acceptable. The staff requires the FSAR to be updated to maintain consistency with values used in the Technical Specifications.

TSPS 825 Reactor Core Isolation Cooling System Isolation Instrumentation

This problem sheet identified a deficiency in the FSAR regarding the reactor core isolation cooling (RCIC) system instrumentation. The FSAR description does not reflect the same level of detail for valve group isolation as contained in the Technical Specifications. From additional review of the design, the licensee has confirmed that the Technical Specifications correctly reflect the instrumentation logic configuration. Based on its review of the problem sheet and the FSAR and discussions with the licensee, the staff finds that no Technical Specification changes are required. The staff requires the FSAR to be changed to provide a more detailed description of the isolation instrumentation.

16.3.4 Clarification of Apparent Problems

The Technical Specification Problem Sheets described herein were resolved by clarification of apparent inconsistencies or deficiencies. In some cases inconsistencies were insignificant because of their small magnitude. In other cases, an apparent deficiency was resolved by considering requirements in another section of the Technical Specifications. In some other cases, administrative controls or plant procedures were found to provide reasonable assurance of safe plant operation.

TSPS 014 Scram Discharge Volume Level Sensors

This problem sheet identified an apparent deficiency in a surveillance requirement regarding surveillance of the scram discharge volume level instrument channel trip unit, which does not include surveillance of the sensors. From its review, the licensee has determined that other Technical Specification requirements address surveillance testing of the sensors. Based on the review of the problem sheet, discussions with the licensee, and a review of the Technical Specifications, the staff has determined that other surveillance requirements do address surveillance of the sensor and finds that no Technical Specification changes are required.

TSPS 044 Division 3 4.16 kV Bus Undervoltage Trip

This problem sheet identified an apparent need for clarification of Technical Specifications regarding time delays for undervoltage trips on Division 3 bus. The time delay to trip the incoming breaker for loss of power on Division 3 would be bypassed. After discussions with the licensee regarding the potential for spurious trips resulting from momentary inrush current during lightning or switching transients, the licensee agreed to retain the time delay in the Technical Specifications. Therefore, no Technical Specification change is required.

TSPS 056 HPCS Automatic Transfer

This problem sheet discussed the need for a specification to verify operability of the condensate tank to suppression pool transfer system. The staff has reviewed the Technical Specifications and notes that surveillance testing is required in another part of the Technical Specifications to verify operability of the transfer system. No change is therefore required.

TSPS 064 Plant Safety Review Committee (PSRC)

This problem sheet identified an apparent need for further definition of responsibilities for a PSRC alternate member. The Technical Specifications state that the Plant Manager may appoint "alternate members" to the Plant Safety Review Committee (PSRC) on a "temporary basis." The concerns were that "temporary basis" is not defined and the position responsible for appointing alternate members differs from the PSRC Chairman. The combination of the words "alternate" and "temporary" make it clear that alternates are to serve on the PSRC only when the regular member is not available. Also, since the PSRC functions to advise the Plant Manager, there is no inconsistency in having the Plant Manager appoint the alternates. Therefore, no Technical Specification changes are necessary.

TSPS 065 Safety Review Committee (SRC)

This apparent problem is similar to TSPS 064 except that it applies to the Safety Review Committee (SRC) and only to "temporary" and "alternate members." For the same reason discussed under TSPS 064, no Technical Specification change is necessary.

TSPS 066 Instrumentation Applicability Consistency

This problem sheet concerned apparent inconsistencies with operational conditions associated with common instruments listed in the Reactor Protection System Instrumentation Table, the Emergency Core Cooling System Actuation Instrumentation Table, the Radiation Monitoring Instrumentation Table, and the Isolation Actuation Instrumentation Table. Even though instrumentation may be common to all or some of the four tables, the function of the instrumentation is unique for each application. Considering the different functions of the instrumentation, it is not appropriate to require the same operational conditions for common instruments in the four tables. The present operational conditions listed in these tables is correct for the intended function of each instrument. Therefore, no Technical Specification change is required.

TSPS 068 Containment Purge System Operation

This problem sheet identified an apparent problem regarding the limiting condition for operation of the purge system which reads in part, "... either the 20 inch or the 6 inch purge system may be in operation;". The licensee questioned whether this statement should be clarified to indicate that the 20-in. and 6-in. purge system shall not be in use at the same time. Based on discussions with the staff, the licensee agreed that the statement is sufficiently clear to indicate that both purge systems shall not be used at the same time. Therefore, no Technical Specification change is required.

TSPS 082 Standby Service Water System Actuation Circuitry

This problem sheet identified an apparent deficiency in a surveillance requirement that addresses the standby service water system. The apparent deficiency concerned the surveillance testing of the automatic actuation circuitry associated with the standby service water pumps and cooling water fans. From a review of the system design and Technical Specification requirements, the licensee has determined that no Technical Specification changes are required at this time. The licensee has also determined that other Technical Specification requirements address surveillance testing of the automatic actuation circuits. Based on the review of the problem sheet, discussions with the licensee, and a review of the Technical Specifications, the staff finds that other Surveillance Requirements address these circuits and, therefore, no Technical Specification changes are required.

TSPS 146 Control Room Fire Protection Features

This problem sheet identified an inconsistency between the FSAR and the Technical Specifications regarding certain fire protection features for the control room. In the FSAR, the licensee committed to maintain the access door to the concealed ceiling space above the control room locked at all times and to provide Technical Specification requirements that would prohibit work of any kind in the concealed area except during cold shutdown. The Technical Specifications do not contain these requirements. However, access to this area is controlled by a security alarmed door and the access control program limits traffic and controls the use of volatiles and combustibles. The FSAR identifies additional fire protection features for this area. Considering the present administrative controls on the area, the staff has concluded that no Technical Specification changes are required.

TSPS 149 SER Requirements for Technical Specifications

This problem sheet identified an apparent inconsistency between the staff's Safety Evaluation Report issued in September 1981 and the Technical Specifications regarding three instruments: the thermal power monitor, the level 8 water level trip, and the turbine bypass system. Based on its review, the staff finds that no change is required for these systems for reasons given below.

- Surveillance requirements for the time constant in the thermal power monitor are included as a footnote in the Technical Specifications.
- Availability setpoints and surveillance requirements for the level 8 trip are included in the Technical Specifications.
- No credit is taken for the turbine bypass system in accident or transient analyses as discussed in Supplement 4 to the staff's SER (NUREG-0831). No changes to Technical Specifications are therefore required.

TSPS 163 Suppression Pool Level Specification Clarity

This problem sheet suggested that a Technical Specification surveillance requirement regarding suppression pool water level should be an Action Statement rather than a Surveillance Requirement. If so, it should be moved to the Action Statement section from its current location in the Surveillance Requirement section. The staff has reviewed the subject Action Statement and concludes that it is equally well placed with either the Action Statements or the Surveillance Requirements because it is a conditional Surveillance Requirement. No change is therefore required.

TSPS 165 Main Steam Isolation Valve Leakage Control System Pressure Monitor

This problem sheet identified a potential deficiency in a Surveillance Requirement. Surveillance Requirement 4.6.1.4.d provides requirements for surveillance testing of the main steam isolation valve (MSIV) leakage control system instrumentation. The MSIV leakage control system pressure monitoring instrumentation is exposed to main steam line pressure during normal plant operations. Because this pressure is above the instrument range, these instruments are normally pegged at the high range of the scale. The channel check required by the specification will detect an instrument failure in a mode that results in a low reading, but may not detect an instrument failure in a mode that results in a high reading. From its review, the licensee has determined that other Technical Specification requirements provide surveillance tests that will detect other possible instrument failure modes. In discussions with the staff, the licensee provided a technical justification to confirm that operation with the instrument pegged high will not be detrimental to the components in the instrument loop. Based on the review of the problem sheet, discussions with the licensee, and a review of the Technical Specifications, the staff finds that other Surveillance Requirements in the Technical Specification address methods to detect other failure modes of the instrumentation, and, therefore, no Technical Specification changes are required.

TSPS 178 Motor-Operated Valve (MOV) Thermal Overload Protection Specification Clarity

This problem sheet discussed an apparent awkward wording of this Technical Specification and its associated Action Statement. The staff has reviewed the specification and finds it is sufficiently clear for accurate interpretation and is in compliance with the BWR Standard Technical Specification (NUREG-0123, Revision 3). Therefore, no change to the Technical Specification is required.

TSPS 186 Response to TMI Action Requirements

This problem sheet identified an apparent need to review licensee's responses to ten TMI action items as addressed in IE Bulletin 79-08. The licensee responded to this bulletin in a letter dated March 19, 1980, which properly responded to these items as related to the Technical Specifications. It has been determined that all necessary changes to the Technical Specifications have already been made to adequately address these issues. Therefore, no Technical Specification changes are required for resolution of this problem sheet.

TSPS 199 Scram Discharge Volume Level Trip Bypass

This problem sheet identified an apparent deficiency in the Technical Specifications regarding the scram discharge volume level trip bypass. Although the Grand Gulf design includes a scram discharge volume level trip bypass, this feature is not included in the Technical Specifications. From additional review, the licensee has determined that the bypass performs no safety function and, therefore, it does not need to be included in the Technical Specifications. The scram discharge volume level trip is required to be operable in Operational Conditions 1, 2, and 5. If it is bypassed (either manually via bypass switch activation or through bypass switch failure) in Operational Conditions 1, 2, and 5, subsequent operation must be in compliance with the Action Statement of the Technical Specifications. Although the Draft BWR/6 Standard Technical Specifications include this bypass switch, there is no apparent increase in safety by requiring this feature to be included in the Grand Gulf Technical Specifications.

Draft BWR/6 Standard Technical Specifications have not been approved by the staff. Accordingly, the staff finds that no Technical Specification changes are required.

TSPS 211 Isolation Actuation Instrumentation Specifification Clarity

This problem sheet identified an apparent need for clarification in the Technical Specifications regarding the notation that describes the logic configuration required to initiate a trip. From additional review of the system design, the licensee has determined that the notation could be clarified; however, such changes would not alter any Technical Specification requirements. Consequently, the licensee has determined that no Technical Specification changes are required. Based on a review of the problem sheet and discussions with the licensee, the staff finds that no Technical Specification changes are required.

TSPS 222 Control Room Emergency Filtration Surveillance

This problem sheet identified an apparent deficiency in Technical Specifications regarding the control room emergency filtration system. The NRC staff commented that surveillance requirements should include a requirement to verify that the control room air temperature is ≤120°F at least once every 12 hours. In its review, licensee found that this requirement is covered by a more stringent Surveillance Requirement specifying that the control room temperature must be verified to be below 77°F at least once every 12 hours. Although the licensee has proposed by TSPS 100 to change this temperature limit to 90°F, this is still well below 120°F. Therefore, no Technical Specification change is required.

TSPS 224 Halon Fire Protection System Specification Clarity

This problem sheet identified an apparent deficiency in a Technical Specification Surveillance Requirement concerning halon storage tank weight and pressure acceptance criteria. Based on discussions with the licensee, the staff concluded that because the acceptance criteria (i.e., 95% of full charge weight and 90% of full charge pressure) were clearly stated in the associated limiting condition for operation, there is no need to restate these criteria in the Surveillance Requirement. Therefore, no change to the Technical Specifications is required.

TSPS 226 Reactor Protection System (RPS) Electric Power Monitoring Assembly

This problem sheet questioned the need for an Action Statement requiring action within 30 minutes if both monitoring assemblies are inoperable because another Action Statement requires action within 72 hours with one monitoring assembly inoperable. Based on its discussions with the licensee, the staff concluded that both are required. Therefore, no changes to the Technical Specifications are required.

TSPS 231 Containment Penetrations

This problem sheet identified an apparent need for clarification of a Technical Specification regarding inoperable automatic isolation valves. The specifications require that containment penetrations not capable of being closed by operable containment isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position. It was suggested that "secured in position" be changed to read "secured closed." Based on its review, the staff finds that the present specification statement is sufficiently clear because it is obvious they must be closed. Therefore, no change to the Technical Specification is required.

TSPS 232 Containment Isolation Valve Lists

This problem sheet discussed a suggested change to Technical Specifications to improve clarity of interpretation. The Technical Specification for isolation valves contains surveillance testing requirements for automatic isolation valves. The specification table lists automatic isolation valves in one section, manual valves in another section, and nonelectrically actuated valves in another section. This problem sheet suggests that the manual and nonelectrically operated valves be removed from the table. However, the manual and nonelectrically operated valves are referred to in other areas of the Technical Specifications, such as leak testing surveillance requirements. The table is a compilation of all the containment and drywell isolation valves. Therefore the staff finds that the present Technical Specification table is necessary.

TSPS 242 Average Power Range Monitor Setpoint Bases

This problem sheet identified an apparent discrepency between thermal power contained in a table in the Bases and the rated power level. The nominal thermal power used in Bases Table 2.1.2-2 is 3323 MWT and the Grand Gulf rated thermal power is 3833 MWT. This table lists the parameters used in a statistical analysis for the safety limit MCPR value. This analysis is a bounding generic analysis that applies to the full range of BWR types and core sizes. It is described in GESSAR II (NEDO-24011-P-A-6), which has been reviewed and approved by the staff for application to initial and reload cores. Table 5.2-3b of GESSAR II specifically lists Grand Gulf as one of the reactors to which it applies.

The staff concludes that no change to the Technical Specification Bases is required.

TSPS 246 Fire Protection Sprinkler System Operability

This problem sheet discussed the apparent need to revise a Technical Specification Surveillance Requirement regarding the operability of wet pipe sprinkler systems. The Technical Specifications require the demonstration of sprinkler system operability "by performing a system functional test which includes simulated automatic actuation of the system...." The licensee believed that control building sprinkler systems had no components that were capable of automatic actuation and that, accordingly, this Surveillance Requirement was not applicable to these systems and that a modification to the Technical Specifications to indicate this was not necessary. However, there is an alarm that is received in the control room initiated by a flow switch installed in these systems whenever flow is initiated in the systems. This function will be tested in accordance with the Technical Specifications. The staff concludes that this requirement is clear in the Technical Specifications as presently written. Therefore, no change to the Technical Specifications is required.

TSPS 247 Fire Protection Halon System Surveillance Requirement Clarity

This problem sheet discussed an apparent need for clarif' .tion of a Technical Specification Surveillance Requirement regarding the h system, including associated ventilation system fire damper logic. Th .nnical Specification Surveillance Requirement requires verification that the halon systems, including associated ventilation system fire damper logic, activate automatically on receipt of a simulated activation signal. The halon systems for the subfloor areas do not have an "associated ventilation system" and, therefore, have no fire dampers. The licensee questioned whether there is a need to change the Technical Specifications to delete "associated ventilation system fire damper logic" from the applicable Surveillance Requirement. The staff concluded that because the term "associated" is used, the surveillance only requires a check of fire damper logic if it exits. Furthermore, the Surveillance Requirement, as written, requires an automatic activation check for halon systems without "associated" ventilation systems. Therefore, no change to the Technical Specifications is required.

TSPS 250 Standly Ligquid Control System (SLCS) Operational Conditions

This problem sheet raised the question as to whether the SLCS should be required to be operable in Hot Shutdown and Cold Shutdown because single control rods can be moved in these operational conditions. The current Technical Specifications do not require the SLCS in these operating conditions. Based on its review, the staff concludes that because there is sufficient shutdown margin to maintain shutdown with the highest worth rod stuck out, the SLCS is not needed in the operational conditions. The staff concludes that no changes to the Technical Specifications are required.

TSPS 254 Startup Channel Check Requirements Clarity

This Technical Specification problem sheet identified an apparent need for clarification of a Technical Specification concerning the table notation applicability for checks of startup channels. From additional review of the Technical Specification requirements, the licensee has determined that the table notation is acceptable and no Technical Specification changes are required. Based on its review of the problem sheet and discussions with the licensee, the staff finds that no Technical Specification changes are required.

TSPS 261 Scram Accumulator Operability

This problem sheet identified a concern with respect to the operability of the scram accumulators with a degraded control rod drive (CRD) pump. The moving of a control rod one notch to ensure a control rod drive pump is operating does not verify that there is sufficient pressure available to the remaining accumulators. Based on its review, the staff concludes that with the control rods out, the reactor will be at power and the accumulators are not required to scram the reactor. Although the accumulators provide a "head start" for the rods, reactor pressure alone is capable of scramming the reactor. Thus, no change to the Technical Specifications is required.

TSPS 263 Turbine Stop Valve and Control Valve Position Sensors

This problem sheet identified an apparent inconsistency between two Technical Specification Action Statements that prescribe requirements applicable when the turbine stop valve and turbine control valve position sensing instrumentation is inoperable. The apparent inconsistency concerns dissimilar requirements for those portions of the instrumentation common to the reactor protection system and the recirculation pump trip system. From its review, the licensee determined that more stringent requirements are appropriate for the reactor protection system instrumentation. From additional review of the systems designs, Technical Specification requirements and functional requirements of each system, the licensee has determined that no Technical Specification changes are required. Based on the review of the problem sheet, discussions with the licensee, and a review of the Technical Specifications, the staff finds that no Technical Specification changes are required.

TSPS 264 Main Steam Isolation Valve Operability

This problem sheet concerned an apparent inconsistency between two Technical Specifications concerning the times to take action when a main steam isolation valve (MSIV) is inoperable. The times to take action are different because of operational and safety considerations on which each specification is based. In the event an MSIV becomes inoperable, the operator will take the required action of the applicable specification. It is not necessary to use the same time for both specifications or to cross-reference the two specifications. Therefore, no Technical Specification change is required.

TSPS 270 Program To Reduce Radioactive Water Leakage Outside Containment

This problem sheet identified an apparent deficiency in Technical Specification requirements for this program. Technical Specification 6.8.1 does not currently require a written procedure to meet the requirements of NUREG-0737, Item III.D.1.1. However, Technical Specification 6.8.3.a requires a program to reduce radioactive leakage outside the containment to be "established, implemented and maintained." The program described in Technical Specification 6.8.3.a adequately addresses the program requirements that were accepted by the NRC staff in the SER. Therefore, no change to the Technical Specification is required.

TSPS 271 Halon Fire Extinguishing System Pressure and Weight Requirements

This problem sheet described an apparent discrepancy between the design capability and the Technical Specification requirement for the correct pressure and weight to be maintained by a halon fire extinguishing system. The Technical Specifications require the halon storage tanks to have at least 95% of full charge weight and 90% of full charge pressure. A preliminary review by the licensee of system design requirements and preoperational test reports indicated that 95% of full charge weight was not sufficient to provide a flooding concentration. A design change was considered to increase the charge weight per cylinder to ensure that the 95% criterion in the Technical Specifications would provide a minimum halon concentration of 5% at 10 minutes after discharge within the protected area. Final design evaluation by the licensee, however, found that the original design (i.e., charge weight per cylinder of 174 pounds) is sufficient to satisfy system design requirements and the 95% criterion ensures the necessary flooding concentration. Therefore, no change to the design or Technical Specifications is required.

TSPS 273 Automatic Depressurization System (ADS) Action Statement

This problem sheet indicated that an Action Statement is needed to address the condition when an ADS trip system is inoperable. Based on a review of the relevant Technical Specifications, the staff found that an applicable action is already defined in another part of the specifications. Therefore, no change to the Technical Specification is required.

TSPS 274 High Pressure Core Spray (HPCS) Action Statements

This problem sheet identified an apparent need for clarification of the applicability of certain Action Statements and notations for the HPCS system and its associated diesel generator. Based on its review, the staff concludes that no change to the Technical Specification is required.

TSPS 279 ECCS Instrumentation Response Times

This problem sheet identified an apparent deficiency in Technical Specification 3.3.3 regarding the requirements applicable when the emergency core cooling system instrumentation exceeds the prescribed response times. From additional review of the Technical Specifications and from discussions with the NRC staff, the licensee has determined that a present Technical Specification Action Statement will apply when response times are outside the specified limits. Based on the review of the problem sheet, discussions with the licensee, and a review of the Technical Specifications, the staff finds that no Technical Specification changes are required.

TSPS 284 Radioactive Gaseous Effluent Monitoring Instrumentation

This problem sheet identified an inconsistency between the Draft BWR/6 Standard Technical Specifications (STS) and a Technical Specification that lists the requirements for radioactive gaseous effluent monitoring instrumentation. The surveillance frequencies for channel checks and channel functional tests for the offgas pre-treatment and post-treatment monitors are not consistent with the frequencies in the Draft BWR/6 STS. The NRC staff has not approved the Draft BWR/6 STS.

The surveillance frequencies for the offgas pre-treatment and post-treatment monitors given in the Grand Gulf Technical Specifications are consistent with the requirements in the Standard Radiological Effluent Technical Specifications for Pressurized Water Reactors (NUREG-0473, Revision 3, January 1983), which is the applicable regulatory guideline for these specifications. However, to avoid any possibility of misinterpretation of conflicting action statements, the surveillance requirements for these monitors were deleted from the Grand Gulf Technical Specifications (see TSPS 120). This deletion did not relax the radiation effluent monitoring instrumentation requirements. Therefore, the staff concludes that no Technical Specification change is required.

TSPS 288 DC Power Source Action Statement Clarity

This problem sheet described an apparent lack of clarity in Technical Specification Action Statements when the Division 3 battery is found to be inoperable. After discussions with the licensee, the staff concluded that proper actions will be taken with the present Action Statements, considering the definition of operability in the Technical Specification. Therefore, no change to the Technical Specifications is required.

TSPS 295 Safety Review Committee Audits Clarity

This problem sheet identified inconsistencies between the wording of the Grand Gulf Technical Specification for audits, and the wording of the General Electric Standard Technical Specifications (GE STS), which may indicate a deficiency in the Grand Gulf Technical Specifications. The GE STS have not been approved by the staff. The specific example noted was the absence of the requirement that "qualified licensee QA personnel" perform the 24-month audit of the Fire Protection Program and implementing procedures. The Grand Gulf Technical Specification regarding audits has been reviewed and represents an acceptable statement of the subjects to be covered by audits. The lack of specification that qualified licensee Quality Assurance (QA) personnel must perform the audit of the Fire Protection Program and implementing procedures is acceptable because alternatives, such as the use of qualified contract personnel, are also acceptable. The lack of specification of who must perform the audit is also consistent with the statements regarding similar program and procedure audits. When performance of the audit is restricted to a specified type or group of qualified auditor (e.g., outside qualified fire consultant), this restriction is indicated in the Technical Specification. Therefore, no changes to the current Technical Specifications are necessary.

TSPS 296 and 341 Plant Safety Review Committee Responsibilities Grand Gulf Nuclear Station

Inconsistencies were identified between the Grand Gulf Technical Specification requirements associated with the Plant Safety Review Committee (PSRC) responsibilities and the PSRC responsibilities as identified in the GE Standard Technical Specifications (GE STS). The GE STS have not been approved by the staff. An organizational unit has been established at Grand Gulf for the review of procedures, modifications, tests, experiments, and violations. In this organizational unit, many of the PSRC reviews are tied to a determination that the subject activity may involve an unreviewed safety question, as defined in 10 CFR 50.59. The responsibility for making that determination is assigned to a technical review and control group of the plant staff. The responsibilities and qualification requirements for this group are detailed in Grand Gulf Technical Specifications for which there is no corresponding section in the GE STS. The staff has reviewed the combined responsibilities of the PSRC and the technical review and control group and concludes that all necessary reviews have been specified. Furthermore, the staff concludes, based on its review of the technical review and control group requirements, that the reviews by this group will be performed by personnel with the necessary independence and technical expertise. Therefore, the staff concludes that the PSRC requirements, as specified in the GGNS Technical Specifications, are acceptable and no Technical Specification changes are needed.

TSPS 297 Reactor Vessel and Recirculation System Water and Steam Temperatures

This problem sheet discussed an apparent inconsistency between the nominal average reactor coolant temperatures in the design section of the Technical Specifications and the temperatures shown in the FSAR. The average temperature (T_{ave}) at rated power conditions in the FSAR is used in reactor performance calculations.

The staff has reviewed the apparent inconsistency and notes that the average temperature in the design section of Technical Specifications is used to calculate reactor vessel and coolant pipe expansion in computing reactor coolant system volume. The T_{ave} given in specifications, therefore, may not correspond exactly to the average temperature at rated conditions in the FSAR. However, it need not be the same, since differences of 25°F will have an insignificant effect on the estimate of reactor coolant system volume. No change to the Technical Specification is, therefore, required.

TSPS 301 Technical Review of Safety-Related Activities

This problem sheet described a concern that the qualification requirements for personnel performing reviews of nuclear safety-related activities were overly restrictive. The Technical Specification states that individuals responsible for the subject reviews shall meet or exceed the qualification requirements of Section 4.4 of ANSI N18.1-1971. The licensee felt that this could be interpreted to mean that one individual must meet all the requirements of Section 4.4 of the standard even if none of the technical disciplines discussed in that section were applicable to that review. That section of the standard requires a certain level of qualification for professional-technical groups and specifies the minimum requirements in four disciplines. Because the standard specifies a minimum level of staff qualification and does not require that all the qualifications be satisfied by one individual, the Technical Specification is understood to require that reviews of safety-related activities will be performed by personnel with a level of qualification equivalent to or as specified in ANSI N18.1-1971, Section 4.4, and that that qualification will be in the discipline(s) applicable to the subject safety-related activities. Based on the above, no change to the Technical Specifications is required.

TSPS 314 Average Power Range Monitor Operability

This problem sheet identified an inconsistency between the GE Standard Technical Specifications (GE STS) and the Technical Specifications regarding the average power range monitor (APRM) operability requirements. The inconsistency concerns the applicability of Operational Condition 4 for APRM operability. The NRC staff has not approved the GE STS. From additional review of the reactor protection system functional requirements and the accident analysis for Grand Gulf, the licensee has determined that because the reactor is subcritical in Operational Condition 4, no Technical Specification changes are required. Based on its review of the problem sheet and discussions with the licensee, the staff finds that no Technical Specification changes are required.

TSPS 324 Shift Staffing Requirements

This problem sheet identified an apparent discrepancy between shift staffing requirements of 10 CFR 50.54(m) and requirements specified in the Technical Specifications. Technical Specifications specify unit operating staff requirements, minimum shift crew compositions, and additional restrictions regarding unexpected staff absence and shift relief. The NRC regulations present shift staffing requirements associated with senior reactor operators and reactor operators. Although the wording of the requirements in the NRC regulations differs from that in the Technical Specifications, all staffing requirements specified in 10 CFR 50.54(m) are incorporated in the Technical Specifications. Therefore, no change in the Technical Specification is required.

TSPS 335 Fuel Oil Sampling and Analysis

This problem sheet described apparent discrepancies between the Technical Specifications and NRC Regulatory Guide 1.137; "Fuel Oil Systems for Standly Diesel Generators," Revision 1, October 1979. These apparent discrepancies were discussed with the licensee, who would like clarifications added to the Technical Specifications. Based on its review, the staff concludes that present Technical Specifications are in conformance with RG 1.137. Therefore, no change to the Technical Specifications is required.

TSPS 336 Crane Operations Over Spent Fuel Pool

This problem sheet identified an apparent inconsistency between two Technical Specification action statements regarding crane operations over spent fuel storage pool when power supplies are inoperable.

One Action Statement pertaining to "A.C. Sources - Shutdown" includes a requirement for suspending crane operations over the spend fuel storage pool. Another Action Statement pertaining to "Electrical Power Systems Distribution - Shutdown" does not contain a similar requirement. A possible change was identified by the licensee to make both Action Statements include the requirement. A review of the two specifications indicated, however, that an addition to Technical Specifications to require suspension of crane operations was not necessary. It is appropriate to suspend crane operations under circumstances in which facility ac power sources (both onsite and offsite) would have reached a significant level of degradation, which could result in a complete loss of safety functions of critical systems that might be needed to mitigate the consequences of an accident involving crane operations. A similar action statement for the Technical Specification pertaining to electrical power system distributions is not necessary because only individual load centers and motor control centers would be potentially deenergized, affecting only individual components. This condition is less severe than facility onsite/offsite power source degradation and does not require discontinuing crane operations because critical system safety functions would not necessarily be affected. Action statements for individual systems affected would be applicable and they provide adequate assurance of safety in this case. Therefore, no change to the Technical Specifications is required.

TSPS 337 Diesel Generator Operability Requirements Clarity

This problem sheet identified an apparent need for clarification of operability requirements in the Shutdown Condition for the diesel generators and power distribution systems. Based on its review, the staff concludes that the definition of operability in the Technical Specifications makes it clear that Division 1 power distribution systems cannot be declared operable if Division 1 diesel generator is inoperable. Therefore, no change to Technical Specifications is required.

TSPS 343 Battery Low Voltage Monitor

This problem sheet identified a difference between the draft BWR/6 STS and the Technical Specifications regarding the monitoring of the battery low voltage. The staff has not approved the Draft BWR/6 STS. The Technical Specifications comply with the BWR Standard Technical Specifications (NUREG-0123, Revision 3) and are therefore acceptable. No change to the Technical Specifications is required.

TSPS 353 Diesel Generator Fuel Storage Requirements

This problem sheet identified a suggestion to require 48,000 gallons of fuel for each "operable" diesel generator rather than for each diesel generator. Based on its review, the licensee concluded that no change was required. The staff concludes that no change to the Technical Specifications is required.

TSPS 358 Radioactive Gaseous Effluent Monitoring Instrumentation

This problem sheet described the procedure and equipment used to implement an Action Statement and questioned whether the Technical Specification should be made more explicit in accordance with plant practices.

The Action Statement of the Technical Specification for the sampler flow rate measuring device requires that the flow rate be estimated at least once every 8 hours. This action is taken when the number of operable channels is less than the minimum number of operable channels required. The current plant operation will use the auxiliary sampling equipment (which includes a flow rate indicator) to monitor the releases of radioactive materials in gaseous effluents to satisfy the action statement requirement. This auxiliary sampling equipment, which includes a flow rate indicator for measuring sample flow rate, satisfies the Action Statement requirement in the Technical Specification. The use of the auxiliary sampling equipment does not relax a radioactive gaseous effluent monitoring requirement and, therefore, no Technical Specification changes are required.

TSPS 359 Containment Spray System Initiation Instrumentation

This problem sheet identified a potential deficiency in the Technical Specifications regarding the setpoint for containment spray system initiation. The potential deficiency concerned premature initiation of the spray system if the setpoint is below the established value, as permitted by the Technical Specification. From additional review of the spray system functional requirements and spray system design, the licensee has determined that no Technical Specification changes are required. FSAR Section 6.5.2.2 states that containment spray may be initiated regardless of containment pressure to suppress airborne radiation levels in a post-LOCA environment. In addition, timers are provided in the initiation circuits to prevent premature low-pressure coolant injection (LPCI) flow diversion to containment spray. Based on the review of the problem sheet, discussions with the licensee, and a review of the system design, the staff finds that no Technical Specification changes are required.

TSPS 365 Battery Performance Tests

This problem sheet identified apparent inconsistencies between industry standards, regulatory guides, and the Technical Specifications regarding battery performance and service tests. Based on the staff's review, the staff concludes that the Technical Specifications meet the requirements in the approved BWR Standard Technical Specifications (NUREG-0123, Rev. 3). Therefore, no change to the Technical Specifications are required.

TSPS 378 Standby Gas Treatment System Flow Limits

This problem sheet concerns an apparent discrepancy within Technical Specification Surveillance Requirements. One Surveillance Requirement requires a standby gas treatment system (SGTS) flow rate not exceeding 4000 cfm during verification of secondary containment integrity, while another Surveillance Requirement requires that an SGTS flow rate of 4000 cfm ± 10% be verified during testing of the SGTS for operability. However, after further evaluation and discussion with the licensee, the staff agrees that no discrepancy exists. The limit of 4000 cfm is conservative with respect to the analytical limit for the SGTS of 4500 cfm. The limit of 4000 cfm ± 10% is appropriate for high efficiency particulate air (HEPA) filter efficiency testing. Therefore, no Technical Specification change is required.

TSPS 381 Instrumentation Terminology

This problem sheet identified a potential deficiency in the Technical Specifications regarding the use of the term "redundant channel" and the phrase "monitoring that parameter." From additional reviewof the system design and Technical Specification requirements, the licensee has determined that no Technical Specification changes are required. Based on its review of the problem sheet and discussions with the licensee, the staff finds that no changes are required.

TSPS 803 Radiation Monitor Instrumentation Requirement Clarity

This problem sheet identified several apparent inconsistencies between the FSAR and the Technical Specifications. Technical Specifications regarding radiation monitoring instrumentation and the FSAR descriptions of such instrumentation are not identical as follows:

(1) The FSAR contains the following information that is not addressed in the Technical Specifications: (a) detector type; (b) sample line or detector location; (c) scale; (d) purpose of measurement; and (e) principal radionuclides detected.

(2) The Technical Specifications make reference to the minimum channels Operable and the FSAR addresses the number of channels.

(3) The FSAR tabulation of process and effluent radiation monitors addresses radiation monitors for the main steam line and liquid radwaste effluent. The Technical Specification of radiation monitoring instrumentation does not address these items.

(4) The FSAR references several GE systems and microprocessor systems used for monitoring the containment vent, offgas and radwaste building vent, fuel handling area vent, and turbine building vent. The Technical Specifications do not specifically identify the different types of monitoring systems used.

(5) The Technical Specifications address area monitors for the fuel handling area and the control room. The FSAR Tabulation of Process and Effluent Radiation Monitors does not address area monitors.

The staff reviewed these apparent inconsistencies and found that:

(1) The system design information is not required to be in the Technical Specifications.

(2) The number of radiation instrument channels given in the FSAR and the number of minimum operable channels given in the Technical Specification are not the same and need not be identical.

(3) The radiation monitors for the main steam line and liquid radwaste effluent are listed in Technical Specifications regarding isolation actuation instrumentation and Technical Specifications regarding radioactive liquid effluent monitoring instrumentation, respectively.

(4) Description of radiation monitors are not required to be in the Technical Specifications.

(5) The area monitors for the fuel handling area and the control room are listed in a different table of the FSAR.

Based on it review, the staff concludes that no Technical Specification changes or FSAR revisions are required.

TSPS 804 Diesel Generator Fuel Oil Tank Capacity

This problem sheet identified an apparent inconsistency between two sections of the FSAR regarding the required oil capacity of the day tank. One FSAR section states the day tank low level alarm annunciates when 30 minutes of fuel are left. Another FSAR section states the day tank has a capability equivalent to 2 hours of operation (approximately 220 gallons of fuel).

The day tank low level alarm is not associated with the tank's 2-hour required capacity. Therefore, neither the FSAR nor the Technical Specification requirement for a minimum of 220 gallons needs to be changed.

TSPS 809 FSAR Description of MOV Thermal Overload Bypass Circuitry

This problem sheet suggested the description of all three methods of wiring used in the motor-operated valve (MOV) thermal overload bypass circuitry be included in the FSAR. FSAR Amendment 58 revised the applicable section to more clearly describe this circuitry.

Thermal overload bypass circuitry of MOV in the Technical Specification is in accordance with the BWR Standard Technical Specifications and, therefore, no Technical Specification change is required.

TSPS 811 and 813 Plant Safety Review Committee Requirements

This problem sheet described an apparent problem in that not all Plant Safety Review Committee (PSRC) requirements identified in the Technical Specifications are also identified in the Operational Quality Assurance Manual (OQAM). Similarly, the requirements for the Manager of QA, as shown in the OQAM, do not include that the person in that position shall be a member of the Safety Review Committee as required by Technical Specifications. The lack of some details in the OQAM does not relieve the licensee from meeting the total of its requirements and particularly those in the Technical Specifications. Therefore, no change to Technical Specifications is required.

TSPS 814 Safety Review Committee Requirements

An inconsistency was identified between the titles of personnel who compose the Safety Review Committee (SRC), as specified in the Technical Specifications, and those titles as specified in the staff's SER. The difference in titles results from title changes made by the licensee since the SER was written. The staff has reviewed the current Technical Specification regarding SRC composition and concludes it requires an acceptable level of competence to perform the SRC functions. An SER update as a result of title changes is not necessary, nor is any Technical Specification change required.

TSPS 815 Reactor Coolant Chemistry

This problem sheet identified a discrepancy between (1) time required for shutdown if pH exceeds limits, and (2) frequency of in-line calibration for the continuous conductivity monitor given in the Technical Specifications and those provided in the FSAR.

The FSAR contains typical BWR reactor coolant system chemical requirements that are not plant specific for Grand Gulf. The Technical Specification meets regulatory requirements. Therefore, no change to the Technical Specification is required.

TSPS 821 Control Rod Drive Accumulator Level

This problem sheet identified a difference between the FSAR and the Technical Specifications regarding the control rod drive (CRD) level indication. The FSAR indicates that the CRD accumulator pressure and level will be verified weekly. Like most BWRs, Grand Gulf does not have instrumentation to indicate accumulator level, only a high level alarm. The accumulators are not required during normal operation for a reactor scram. Furthermore, the loss of an accumulator at lower pressure will only reduce the speed of rod insertion. Assuming the loss of the ability to insert a rod due to the loss of an accumulator, the plant is designed for safe cold shutdown with the most reactive rod fully withdrawn. Based on the above, the staff concludes that level indication of the water in the accumulators is not necessary. Therefore, no change to the Technical Specifications is required.

TSPS 824 Containment and Drywell Isolation Valves

This problem sheet described an apparent inconsistency between the FSAR and the Technical Specifications. The specifications list several containment and drywell isolation valves that are not listed in the FSAR tabulation pertaining to containment isolation valves.

The Technical Specifications tabulation pertains exclusively to drywell isolation valves (i.e., to maintain drywell integrity), whereas the FSAR tabulation pertains exclusively to containment isolation valves that focus on the containment boundary. The staff finds the Technical Specifications tabulation to be acceptable, even though it does not require the drywell isolation valves to be listed. The regulations that apply to the containment isolation valves as tabulated in the FSAR are not valid for the drywell isolation valves. Therefore, it would be inappropriate to add drywell isolation valves to the FSAR tabulation. Therefore, the staff concludes that neither Technical Specification changes nor FSAR changes are required.

TSPS 827 Fire Suppression Water System Pressure

This problem sheet described an apparent inconsistency between the FSAR requirement and Technical Specification the requirements for the fire suppression water supply.

The FSAR gives system pressure as 125 psig in general system description statements, whereas a Technical Specification Surveillance Requirement specifies 120 psig as the minimum system pressure limit. The 120 psig pressure is adequate because only 118 psig is required for the maximum 2717 gpm sprinkler flow plus 1000 gpm for hose streams. Therefore, no change to the Technical Specifications is required. TSPS 827 incorrectly stated that the automatic makeup to the firewater storage tank occurs at 45 in. below the overflow pipe. This value is associated with a design change being considered by the licensee. The current automatic makeup point is 18 in. below the overflow pipe, which is as stated in the FSAR. Therefore, no change to the FSAR is necessary at this time.

TSPS 829 Reactor Coolant System Chemistry Requirements

This problem sheet identified inconsistencies between the FSAR and the Technical Specifications for (1) reactor shutdown time when pH is out of limits and (2) Surveillance Requirement for a channel check of the continuous conductivity monitor with an inline flow cell whenever conductivity exceeds limits.

The FSAR contains typical BWR reactor coolant system chemical requirements that are not plant specific for Grand Gulf. The Grand Gulf Technical Specification on chemistry meets regulatory requirements. Therefore, no change to the Technical Specification is required.

16.3.5 Superseded Problem Sheets

The Technical Specification Problem Sheets described herein were superseded by other problem sheets and Technical Specification changes.

TSPS 027 Tables of Snubber Lists

This problem sheet identified an error in two table numbers that referred to lists of snubbers. The problem was superseded by a requested change to delete the tables from the Technical Specifications (TSPS 006).

TSPS 034 Instrumentation Review

This problem sheet identifies a potential deficiency in the Technical Specifications regarding the minimum operable channels requirements, Action Statements, and definitions of channel, trip system, and trip functions. Although Technical Specification changes either have been made or will be made as a result of the licensee's resolution of the identified deficiency, the specific changes are addressed by other problem sheets. Based on its review of the problem sheet and discussions with the licensee, the staff finds that no Technical Specification changes are required.

TSPS 059 Embankment Slope Stability

This problem sheet identified a need to clarify the specification for surveillance of a culvert designed to pass flood waters through it. This problem sheet is superseded by TSPS 133.

TSPS 080 Containment Spray Actuation Instrumentation

This problem sheet was a duplicate of TSPS 054, which was resolved by issuing a Technical Specification change in the NRR Director "Order Restricting Condition for Operation," dated April 18, 1984.

TSPS 140 Isolation Valve Closure Time

This problem sheet identified inconsistencies between the closure times of various values listed in the Technical Specifications. An August 9, 1983, request for Technical Specification changes was withdrawn by the licensee's letter dated April 17, 1984. The problem is now identified in TSPS 306.

TSPS 147 Class IE Bus Undervoltage Protection

This problem sheet identified a need to review Technical Specifications to determine conformance to staff's SER requirements regarding Division 1, 2, and 3 electrical buses second-level voltage protection. Based on its review, the staff finds that Surveillance Requirements for second-level undervoltage protection for Divisions 1 and 2 comply with staff requirements. Division 3 undervoltage protection design, which does not comply with staff's requirements, is considered in TSPS 373.

TSPS 188 Scram Discharge Volume (SDV) Vent and Drain Valves

This problem sheet described the resolution of changes to Technical Specifications regarding the SDV as required by NRC Generic Letter dated July 7, 1980. The letter required surveillance requirements for the SDV vent and drain valves and Limiting Conditions for Operation and Surveillance Requirements for the SDV rod block limit switches.

All the required changes have been incorporated into the Technical Specifications except for the SDV trip bypass, which is covered in TSPS 199.

TSPS 212 Instrument System Trips

This problem sheet identified a potential Technical Specification change that would allow inoperable channels to be placed in the tripped condition, rather than the trip system. This problem sheet is a duplicate of TSPS 112.

TSPS 215 Thermal Power Time Constant Surveillance

This problem sheet identified a potential deficiency regarding surveillance of the Average Power Range Monitor Thermal Power Time Constant Setpoint. This problem sheet is a duplicate of TSPS 149.

TSPS 230 Primary Containment Integrity

This problem sheet is a duplicate of TSPS 144.

TSPS 239 Plant System Actuation Instrumentation

This problem sheet was considered together with TSPS 054, which was evaluated in the April 18, 1984, Order.

TSPS 317 High-Pressure Core Spray (HPCS) Pump Capacity

This problem sheet identified an apparent inconsistency concerning HPCS pump capacity. This problem sheet has been superseded by TSPS 256.

TSPS 318 Reactor Mode Switch

This problem sheet identified a potential deficiency in Technical Specifications regarding the number of operable channels per trip system for the reactor mode switch. Resolution of TSPS 197 addresses the need for Technical Specification changes in this area.

TSPS 328 Containment and Drywell Radiation Monitors

This problem sheet is a duplicate of TSPS 367. Resolution of TSPS 367 addresses the need for Technical Specification changes in this area.

TSPS 331 Reactor Core Isolation Cooling (RCIC) System Pressure Isolation Valve Leakage

This problem sheet concerned the alarm setpoint for the pressure used to indicate excessive intersystem leakage past valves isolating the reactor coolant system from the RCIS. This problem sheet discusses areas covered by TSPS 032.

TSPS 368 Radiation Monitors

This problem sheet identified an editorial change of "monitor" to "radiation monitor." This change was a part of TSPS 329 that was made in the NRR Director's April 18, 1984, Order.

16.4 Evaluation of Technical Specification Changes

The licensee submitted requests for Technical Specification changes by letters dated June 9, 1983, September 9, 1983, June 17, 18, 19, 20, 21, and 22, 1984. The 1983 letters are requests that were pending before the licensee implemented the Technical Specification Review Program (TSRP). The 1984 requests resulted from the TSRP.

The evaluations are summarized in two subsections. Section 16.4.1 summarizes the staff's evaluation of substantive changes made to the Technical Specifications, in the order in which changes appear in the Technical Specifications. Section 16.4.2 lists miscellaneous changes made to correct typographical errors, clarify the text, and incorporate editorial comments. Each Technical Specification section number and title is underscored to clearly identify the changes discussed. In addition, the Technical Specification Problem Sheet (TSPS) number is used to correlate the changes to the licensee's review program.

16.4.1 Substantive Changes

3.1.1 Shutdown Margin (TSPS 124)

Surveillance Requirement 4.1.1.c, page 3/4 1-1

The proposed change would increase the time allowance for verifying adequate shutdown margin in the event of a stuck (immovable) withdrawn control rod from 1 to 12 hours. The increased time would permit the use of offsite computer calculations to determine rod worths.

Such an extended time for shutdown margin surveillance has previously been approved for boiling water reactors (e.g., LaSalle Units 1 and 2) and is acceptable for Grand Gulf.

3.1.2 Reactivity Anomaly and LPRM Calibration (TSPS 265)

Surveillance Requirement 4.1.2.b, page 3/4 1-2

The proposed changes would alter the surveillance interval from 31 effective full-power days to 1000 megawatt days per ton of uranium (MWD/T). This represents an increase of approximately 30% in the interval. However, the mechanisms that would cause the reactivity anomaly to change vary slowly with time, and the increase in surveillance interval does not significantly affect the monitoring of this quantity. The 1000 MWD/T interval coincides with that used for rod interchanges. The staff finds these changes acceptable.

3.1.3.1 Control Rod Operability (TSPS 241)

Action Statements a and b, pages 3/4 1-3, 3/4 1-4

The changes to this specification are proposed to clarify its meaning and add an Action Statement addressing scram discharge volume drain and vent valve operability. The substance of the specification is not altered. The staff concludes that the changes are acceptable.

3.1.3.2 & 3.1.3.3 Control Rod Scram Times and Accumulators (TSPS 108)

Action Statements, pages 3/4 1-7, 3/4 1-8

This change would consist of adding an additional statement to the Action Statement of each specification that reads "The provisions of Technical Specification 3.0.4 are not applicable."

Technical Specification 3.0.4 prohibits entry into an Operational Condition unless the conditions for the Limiting Conditions for Operation can be met without reliance on the provisions contained in the Action Statements of the specification. However, exceptions to this requirement may be made where startup with inoperable equipment would not affect plant safety.

In the particular specifications considered herein, entry into Operational Conditions 1 and 2 would be prohibited by Technical Specification 3.0.4 if there were rods with slow scram time or with inoperable control rod scram accumulators. The proposed change would allow such entry.

The staff concludes that this change to the Technical Specifications is acceptable because the Action Statements would still have to be met.

3.1.3.4 Control Rod Drive Coupling (TSPS 371)

Action Statement, page 3/4 1-10

The change to this specification consists of the insertion of an exemption to the provisions of Technical Specification 3.0.4. The effect of the exemption is to permit entry into Operational Conditions 1 and 2 with a decoupled rod. However, the Action Statements in the specification would still have to be met. These permit operation for only a limited time with a decoupled rod that is not completely inserted. The staff concludes that entry into Operational Conditions 1 or 2 with a decoupled rod is acceptable.

3.1.3.5 Control Rod Position Indication (TSPS 155)

Action Statements, page 3/4 1-12

This proposed change revises Technical Specification 3.1.3.5 to include an exemption to the requirements of Technical Specification 3.0.4 and to add an Action

Statement permitting the intermittent rearming of the control rod to perform testing associated with its restoration to operable status. The same arguments as in Technical Specification 3.1.3.4 apply to the exemption to Technical Specification 3.0.4 and the intermittent rearming of control rods for testing purposes in permitted in other Technical Specifications (e.g., 3.1.3.1 and 3.1.3.4) and is acceptable here. The staff concludes that this change is acceptable.

3.1.4.2 Rod Pattern Control System (TSPS 334)

(1) Action Statements a and b, page 3/4 1-16

One proposed change revises Action Statement a by specifying remedial actions to be taken in the event that the rod pattern control system (RPCS) is inoperable, with thermal power above or below the low power setpoint. Presently, the Technical Specifications provide the same remedial actions; however, the actions are based on being above or below 20% of rated thermal power. The proposed change revises the Action Statement refer to the low power setpoint rather than 20% of rated thermal power. Technical Specification 3.3.6 defines the low power setpoint as 20 + 15 - 0% of rated thermal power. Based on its review, the NRC staff finds that the proposed change provides consistency with the Bases Section of the Technical Specifications, provides more stringent restrictions on control rod movement, and is consistent with the assumptions of the rod withdrawal accident analysis contained in the FSAR. Therefore, the staff finds the proposed change acceptable.

Another proposed change revises Action Statement b to refer to the rod action control system (RACS) rather than the rod gang drive system (RGDS). Action Statement b provides a set of remedial actions should one or more control rod(s) be inoperable. Presently, Action Statement b permits the operator to bypass an inoperable rod within the RGDS. From a review of the Technical Specification and the design details of the rod control and information system (RC&IS), the licensee has determined that the RACS is the appropriate system to be addressed in the Action Statement. Both the RGDS and the RACS are subsystems of RC&IS. In the event of an inoperable control rod, the signal to bypass the control rod is input to the RACS upstream of the RGDS. Based on its review, the NRC staff finds that the proposed change does not alter the existing requirements. The proposed change provides consistency between the design and the Technical Specifications. Therefore, the staff finds the proposed change acceptable.

Another proposed change revises Action Statement b.2.a) to change "control rod" to "control rod(s)." This change corrects a typographical error and is, therefore, acceptable.

(2) Surveillance Requirement 4.1.4.2, page 3/4 1-17

This proposed change would require that the RPCS be surveillance tested after withdrawal of the first in-sequence control rod or gang for each reactor startup. Presently, the Technical Specifications require surveillance testing after withdrawal of the first in-sequence control rod for each reactor startup. The provisions for testing after gang rod withdrawal provide a clarification with respect to the acceptability of gang withdrawal before the RPCS Surveillance, and provide for early testing of the RPCS during the reactor startup in which gang withdrawal is the predominant method. Based on its review, the NRC staff finds the proposed change provides a clarification to the Technical Specifications and a conservative set of requirements for RPCS testing. Therefore, the staff finds the proposed change acceptable.

This proposed change also revises Technical Specifications 4.1.4.2.a and 4.1.4.2.b to indicate that two separate functions of the RPCS are being addressed. This is proposed as a clarification to the Technical Specifications. Based on its review, the NRC staff finds the proposed change adds clarification and does not alter the existing requirements. Therefore, the staff finds the proposed change acceptable.

3.2.1 Average Planar Linear Heat-Generation Rate (TSPS 049)

Surveillance Requirement 4.2.1, pages 3/4 2-1 through 3/4 2-4
Bases, page B 3/4 2-1

The changes to this Technical Specification include the consolidation of the curves of maximum average planar linear heat-generation rate (MAPLHGR) onto a single figure and the inclusion of an exemption to Technical Specification 4.0.4. The first change is editorial in nature and is acceptable. The second change would permit entry into Operational Condition 1 at a power greater than 25% full power without verification that the MAPLHGR limits are met. However this verification must be made within 12 to 24 hours and the rod pattern restrictions below 25% of full power ensure that the limits will be met in this power regime. The staff concludes that this change as well as the accompanying changes in the Bases is acceptable.

3.2.2 Average Power Range Monitor (APRM) Setpoints (TSPS 049)

(1) Action Statement, page 3/4 2-5
Bases, page B 3/4 2-2

In Technical Specification 3.2.2 the time limit for completion of corrective action to restore the setpoints to within allowable values would be increased from the present 2 to 8 hours. Also, the amount by which the APRM gain may be adjusted to account for conditions in which the maximum fraction of limiting power density (MFLPD) is greater than the fraction of rated thermal power (FRTP) would no longer be limited to 10%.

The changes to Technical Specifications 3.2.2 make this specification consistent with that in the latest version of the proposed General Electric (GE) BWR/6 Standard Technical Specification (STS). The new value for the time limit allows 1 hour to be allocated for each APRM channel and permits a careful adjustment to be made. The APRM gain factor may not be used to bring the APRM readings above 100% of full power nor used when the power is greater than 90% of full power. The staff concludes that these changes are acceptable.

(2) Surveillance Requirement 4.2.2, page 3/4 2-5

In Technical Specification 4.2.2 an exemption to the requirements of Technical Specification 4.0.4 would be added. The exemption to Technical Specification 4.0.4 is acceptable for reasons given for Technical Specification 3.2.1 above.

3.2.3 Minimum Critical Power Ratio (TSPS 049, 375)

Surveillance Requirement 4.2.3, page 3/4 2-6
Bases, page B 3/4 2-4, 2-6

This Specification has been changed to make Figures 3.2.3-1 and 3.2.3-2 more legible, to add an exemption to Specification 4.0.4 in Technical Specification 4.2.3, and to provide an expanded discussion of the $MCPR_f$ and $MCPR_p$ in the Bases. The first change is strictly editorial in nature and is acceptable. The second change is acceptable for reasons given in the discussion of the changes to Technical Specification 3/4.2.1 above.

The third change provides the calculational procedures used to establish the $MCPR_f$ and $MCPR_p$ curves. The staff has reviewed the proposed changes. Based on its review, the staff finds that acceptable methods have been used to perform the analysis and appropriate assumptions are made with respect to the events used to establish the $MCPR_f$ and $MCPR_p$ curves. The proposed changes to Bases 3/4.2.3 are therefore acceptable.

3.3.1 Reactor Protection System Instrumentation (TSPS 112, 363)

3.3.2 Isolation Actuation Instrumentation

Action Statements, pages 3/4 3-1, 3/4 3-9

Footnote * that is applied to Action Statement a in Technical Specification 3.3.1 and Action Statement b in Technical Specification 3.3.2 states, "With a design providing only one channel per trip system an inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.2-1 for that Trip Function shall be taken." The proposed change would delete the phrase, "With a design providing only one channel per trip system"...

The Grand Gulf reactor trip system is divided into two trip logics (systems); each trip logic is comprised of one or more instrument channels per monitored variable. The design does not include a one-channel-per-trip-system configuration. Similarly, the isolation actuation system is divided into trip systems. The number of trip systems varies depending on the trip function, with each trip system receiving input from several monitored variables. There are no configurations within the design in which there is only one channel per trip system. Therefore, the staff finds the proposed change acceptable.

Action Statement b of Technical Specification 3.3.2 states, "With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place that trip system in the tripped condition within 1 hour. The provisions of Specification 3.0.4 are not applicable." The proposed change would insert the phrase, "the inoperable channel(s) and/or," between the words "place" and "that" in the Action Statement consistent with the August 31, 1982, version of the BWR/6 STS. The proposed change provides the option of either placing the inoperable channels or the associated trip system in the tripped condition. The wording of Action

Statement b as issued in Grand Gulf Technical Specifications provides a more conservative requirement, initiating isolation, when a channel becomes inoperable. However, placing an inoperable channel in the tripped condition is sufficiently conservative, without initiating isolation, to compensate for the inoperable channel. Therefore, the staff finds that the proposed change is acceptable.

The proposed changes revise footnote ** provided to specify the actions to be taken in the event that instrument channels in more than one trip system are inoperable. The proposed changes reword the existing footnotes with regard to initiating a plant trip if one or more instrument channels are inoperable. Based on its review, the NRC staff finds that the proposed change clarifies the wording of the Technical Specifications. The proposed changes do not alter any of the existing requirements. Therefore, the staff finds the proposed changes acceptable.

3.3.1 Reactor Protection System Instrumentation (TSPS 197, 253, 265, 355, 369)

(1) Limiting Condition for Operation, pages 3/4 3-2, 3/4 3-3

The proposed change revises the minimum number of intermediate range neutron monitoring (IRM) channels required operable by Table 3.3.1-1 from two channels per trip system to three channels per trip system. In Operational Conditions 3 and 4, the IRMs provide reactor protection system input to initiate a reactor trip on an increasing flux level. The system includes four channels per trip system. The design of the IRMs and reactor protection system is such that three operable channels in each trip system provide sufficient monitoring capability and automatic trip logic input. Accordingly, the licensee has proposed to require that one additional channel per trip system be operable consistent with the system design. Another change increases the minimum number of reactor mode switch shutdown position channels required operable from one channel per trip system to two channels per trip system to be consistent with the system design. Based on its review, the NRC staff finds that the proposed changes correct errors in the existing Technical Specifications. The changes provide consistency between Technical Specification Table 3.3.1-1, FSAR Section 7.6.1.5.4, and Bases Section 2.2.1.1. Therefore, the staff finds the proposed changes acceptable.

(2) Surveillance Requirements, pages 3/4 3-7, 3/4 3-8

Footnote (c) to Table 4.3.1.1-1 requires surveillance testing (within 24 hours before startup, if not performed in the previous 7 days) of certain neutron monitoring channels that provide input to the reactor trip system. The licensee has proposed to delete this requirement because other Technical Specifications require these same features to be tested every 7 days. In addition, the licensee has proposed to delete functional test requirements performed before startup for those reactor trip inputs (items 2.b and 2.c) that are not required to be operable before entering Operational Condition 1. These trip inputs will be surveillance tested before entering Operational Condition 1, if they have not been performed in the previous 7 days. Based on its review, the NRC staff finds that the proposed changes provide a clarification of the requirements of the Technical Specifications. Therefore, the staff finds the proposed changes acceptable.

The proposed change also adds footnotes (j) (k) and (l) to Table 4.3.1.1-1 to specify operational modes and plant conditions when Surveillance Requirements are not applicable for certain reactor trip system inputs. The proposed change provides consistency between the Limiting Conditions for Operation and Surveillance Requirements of the reactor trip system instrumentation. Based on its review, the NRC staff finds that the proposed change corrects an error and clarifies the Surveillance Requirements of the Technical Specifications. Therefore, the staff finds the proposed change acceptable.

Footnote (f) to Table 4.3.1.1-1 has been revised to change the surveillance interval for LPRM calibration from every 1000 effective full-power hours to every 1000 MWD/T. Because these are approximately equivalent intervals, this change is acceptable.

3.3.2 Isolation Actuation Instrumentation (TSPS 013, 040, 110, 111, 201, 308, 315, 350)

(1) Limiting Condition for Operation, pages 3/4 3-10, 3/4 3-11, 3/4 3-12, 3/4 3-14, 3/4 3-14a

The proposed change adds a footnote (O) to items 1.c and 1.e of Table 3.3.2-1 Footnote (O) reads as follows: "Also isolates valves E61-F009, E61-F010, E61-F056, and E61-F057 from Valve Group 7." The proposed footnote (O) explains that four isolation valves isolated as a part of Group 7 are also isolated with Group 5 on receipt of a low reactor vessel water level and high drywell pressure signal. Based on its review, the NRC staff finds that the proposed change does not alter the existing requirements. The footnote provides additional information on the Grand Gulf design. Therefore, the staff finds the proposed change acceptable.

The proposed change also deletes the reference to footnote (f) from Table 3.3.2-1. The Grand Gulf design includes features that automatically trip the mechanical vacuum pumps on a main steam line high radiation signal. In addition, the mechanical vacuum pumps can be manually tripped, at any time, should the operator deem it necessary. Footnote (f) on Table 3.3.2-1, Item 3.c, implies that the manual initiation of secondary containment isolation also initiates a trip of the mechanical vacuum pumps. This is not consistent with the design. Accordingly, the licensee has proposed a change to delete the reference to footnote (f). Based on its review, the NRC staff finds that the proposed change corrects an error in the Technical Specifications. Deletion of the reference to footnote (f) does not change the requirements of the Technical Specifications. Therefore, the staff finds the proposed change acceptable.

The proposed change also revised Table 3.3.2-1 Action Statement 28 to provide an option should certain isolation actuation instrumentation channels be inoperable. The option permits continued plant operation provided either the associated isolation valves are lock closed or verified closed and electrically disarmed.

The proposed change revises Table 3.3.2-1 Action Statement 28 to allow alternative methods of ensuring penetration isolation. Presently, Action Statement 28 requires that associated isolation valves be locked closed within 1 hour after determining that the reactor heat removal (RHR) system isolation actuation instrumentation is inoperable. The licensee has stated that this is not possible in the case of the RHR shutdown cooling inboard isolation valve, because this valve is located in the drywell and cannot be locked closed with the reactor at power because of radiological considerations. The proposed change permits, as an alternative, the use of remote indication to verify that the valve is closed. After the valve is verified closed, it is then electrically disarmed. Based on its review, the NRC staff finds that the proposed change provides an acceptable alternative method of ensuring containment isolation should actuation instrumentation become inoperable. Therefore, the staff finds the proposed change acceptable.

This proposed change also adds a time delay for isolation of the reactor core isolation cooling (RCIC) system with a trip setpoint of 5 seconds to Item 5.a of Tables 3.3.2-1 and Item 5.a of 3.3.2-2 and an allowable value of 5 ± 2 seconds to Item 5.a of Table 3.2.2-2. The time delay is required to prevent spurious RCIC isolations that can result from pressure spikes occurring upon system startup. Item 9 of NRC Generic Letter 83-02 requires a delay time sufficient to preclude spurious isolation, but that also allows "true signal" isolations within 7 seconds or less. The time delay setpoint proposed by the licensee, in conjunction with the associated surveillances, satisfies these requirements. Safety is enhanced by the reduction of spurious RCIC system isolations. The NRC staff concludes that the proposed Technical Specification changes are acceptable.

(2) Instrument Setpoints, pages 3/4 3-15, 3/4 3-16, 3/4 3-17, 3/4 3-17a

This proposed change would revise trip setpoints for (1) containment and drywell ventilation exhaust radiation monitor, (2) fuel handling area ventilation exhaust radiation monitor, and (3) fuel handling area pool sweep exhaust radiation monitor. The trip setpoint values that appear in Table 3.3.2-2 (isolation actuation setpoints) are the same values for the alarm setpoints given in Table 3.3.7.1-1 (radiation monitoring instrumentation setpoints). To provide alarm indication before isolation actuation, the licensee has proposed to change the nominal trip setpoints for isolation actuation. By letter dated April 26, 1984, from L. F. Dale (MP&L) to H. Denton (NRC), the licensee stated that (1) there are no accident analyses assumptions or offsite release limits directly associated with these nominal trip setpoint values, (2) the setpoint values were selected to ensure that off-normal conditions will be rapidly detected and isolation will be promptly initiated, and (3) the setpoint values are best-estimate values based on engineering calculations of the full-power background radiation levels at the detector locations plus some amount of radiation indicative of an off-normal condition. As required by the Technical Specifications, the final setpoints will be determined following completion of the startup test program that will include actual measurements of background radiation levels at the detector locations. The licensee has proposed no change to the allowable values (i.e., those radiation levels at which actuation must be initiated). The proposed change is to revise only the nominal trip setpoints. Based on the above, in the interim until the startup test program is completed, the staff finds that the proposed nominal trip setpoints are acceptable.

Another proposed change reduces the allowable value for the residual heat removal/reactor core isolation cooling, steam line flow high isolation function (Item 5.k on Table 3.3.2-2) from less than or equal to 160" H_2O to less than or equal to 151" H_2O. The licensee has stated that the proposed change is based on refined calculations that were performed as a result of the nuclear steam supply system (NSSS) vendor's verification of the design documents. In response to a request from the NRC staff, the licensee is participating in a BWR Owner's Group effort to provide more detailed information on their setpoint methodology. The staff concludes that there is reasonable assurance, based on staff participation in meetings with the BWR Owner's Group working group on setpoint methodology, that the forthcoming more-detailed information setpoints and setpoints methodology being developed by this group will verify the acceptability of the proposed setpoints. In the interim, the staff finds the proposed change acceptable.

Another proposed change to Table 3.3.2-2 is to delete items 4.c.4, 4.c 5, and 4.c.6 (equipment area temperature high) and 4.d.4, 4.d.5, and 4.d.6 (equipment area delta temperature high), trip setpoints and allowable values for the reactor water cleanup (RWCU) demineralizer rooms, receiving tank room, and demineralizer valve room. These equipment areas enclose piping and components that are downstream of the RWCU heat exchangers that contains reactor coolant less then or equal to 120°F during normal operation. Temperature monitoring for leakage in these rooms would not be responsive because of the relatively low temperature of the reactor coolant in these rooms. This Technical Specification, Item 4.a of Table 3.3.2-2, contains a delta flow-high reactor water cleanup system isolation setpoint, which is more sensitive to RWCU system leakage. Temperature monitoring for leakage is not needed and, therefore, deletion of items indicated in attached Table 3.3.2-2, is acceptable.

(3) Instrument Response Times, pages 3/4 3-18, 3/4 3-19

Other proposed changes revise the isolation actuation system response times specified in Table 3.3.2-3 to reflect the design time allowance for the diesel generators to restore ac power on a loss of offsite power. In addition, a response time of ≤ 3 seconds is added to the present requirements for the fuel handling area ventilation and pool sweep exhaust radiation instrumentation, consistent with the assumptions of the accident analysis for a fuel handling accident. Based on its review, the NRC staff finds that the proposed changes provide consistency between the Technical Specifications and the response time assumptions of the accident analysis for Grand Gulf. Therefore, the staff finds the proposed changes acceptable.

(4) Surveillance Requirements, pages 3/4 3-22

This change also adds Surveillance requirements for the instrumentation added to delay isolation of the RCIC system. The RCIC time delay evaluation is presented in paragraph (1) above. The staff finds the proposed surveillance requirements acceptable.

3.3.3 Emergency Core Cooling System Actuation Instrumentation (TSPS 075, 114, 115, 116, 206, 303, 364)

(1) Limiting Condition for Operation, pages 3/4 3-26, 3/4 3-27

One proposed change is to delete footnote e and footnote ** in Table 3.3.3-1.

Footnote e states that "One-out-of-two" is not used in Table 3.3.3-1. If removed from Table 3.3.3-1, there would be no change to the requirements of the Technical Specifications. Therefore, the staff finds the proposed change acceptable.

Footnote **, which is applied to the LOSS OF POWER instrument channels states "Required when Engineered Safety Features (ESF) equipment is required to be OPERABLE," is proposed to be changed as follows: "Required when <u>applicable</u> ESF equipment is required to be OPERABLE." From its review, the staff finds that the proposed change would not change the requirements of the technical specifications. Therefore, the staff finds the proposed change acceptable.

Another proposed change revises Item C.1.f and Action Statement Number 33 to Table 3.3.3-1 to reflect the single trip system design of the high-pressure core spray system. The Grand Gulf design includes features that automatically initiate the high-pressure core spray (HPCS) system. Fifteen instrument channels and one manual initiation circuit are provided within one trip system to initiate HPCS. Presently, the Grand Gulf Technical Specifications are structured for a plant with two trip systems to initiate HPCS. The proposed changes will make the Technical Specifications consistent with the single-trip system design. Based on its review, the NRC staff finds that the proposed change corrects an error in the Technical Specifications. The change does not revise any of the existing requirements. Therefore, the staff finds that the proposed change is acceptable.

(2) Instrumentation Setpoints, pages 3/4 3-28

The licensee has proposed changing the low-pressure coolant injection (LPCI) pump B and C discharge pressure-high from "115 psig, increasing," to "115-135 psig, increasing." The purpose of the lower setpoint limit on this trip is to provide assurance that a low-pressure emergency core cooling system (ECCS) pump is operating before automatic depressurization system (ADS) actuation. The upper setpoint limit ensures that ADS is not prevented from initiating as a result of a setpoint higher than pump capability. Pump characteristic curves in the final safety evaluation report (FSAR) indicate that the 115 to 135 psig range is within the design limits of the LPCI pumps. The proposed change is therefore, acceptable.

Another change to Table 3.3.3-2 is to revise the allowable value for the suppression pool level-high trip that initiates high-pressure core spray and reactor core isolation cooling isolation. For the Grand Gulf design, an elevation of 111 ft 9 3/4 in. corresponds to a suppression pool volume of 138,701 ft^3, the maximum allowable suppression pool volume. This translates to an elevation that corresponds to approximately 7 in. of the range of the level instrument used for transferring HPCS and RCIC suction to the suppression pool. The licensee has stated that a Technical Specification allowable value of 7 in. is consistent with the suppression pool volume requirements and includes allowances for calibration and transmitter inaccuracies to prevent the actual contained volume from exceeding 138,701 ft^3. In response to a request from the NRC staff, the licensee is participating in a BWR Owner's Group effort to provide more detailed information on their setpoint methodology. The final acceptability of the Grand Gulf setpoint methodology, trip setpoints and allowable values will be addressed in a supplement to this report. The staff concludes that there is reasonable assurance, based on staff participation in meetings with the BWR Owner's Group working group on setpoint methodology, that the forthcoming more-detailed information on setpoints and setpoint methodology being developed by this group will verify the acceptability of the proposed setpoints. In the interim, the staff finds that the proposed change is acceptable.

(4) Surveillance Requirements, pages 3/4 3-31, 3/4 3-33

This proposed change would revise surveillance requirements for the automatic depressurization system actuation instrumentation. Footnote a that states "Calibrate trip unit at least once per 31 days" is proposed to be applied to the low-pressure core spray system (LPCS) Pump Discharge Pressure-High instrument channel. This change will make the calibration requirements for this trip unit consistent with those of other Divisions 1, 2, and 3 ECCS actuation instrumentation trip systems. The staff finds that the proposed change provides consistent surveillance requirements for identical instrumentation and is more conservative in that it provides more frequent surveillance. Therefore, the staff finds that the proposed change is acceptable.

Another proposed change deletes footnotes (c) and (d), Note 1, and the reference to footnotes (c) and (d) in items A.1.d and B.1.d from Table 4.3.3.1-1. Footnotes (c) and (d) to Table 4.3.3.1-1 address the surveillance requirements for the low-pressure core spray (LPCS) and low-pressure coolant injection (LPCI) valve interlocks. Note 1 to Table 4.3.3.1-1 states that the requirements of footnotes (c) and (d) are not applicable until restart after the first refueling outage. In accordance with License Condition 2.c (2.1), the LPCS and LPCI interlocks are required to be installed and operable before restart following the first refueling outage. Accordingly, the licensee has proposed to remove all references to these interlocks from the Technical Specifications now, and submit Technical Specification changes to include appropriate requirements prior to implementation of the design change. Based on its review, the NRC staff finds that the proposed change does not alter the existing requirements. Therefore, the staff finds the proposed change acceptable.

3.3.4.1 ATWS Recirculation Pump Trip System (TSPS 022)

Action Statement, pages 3/4 3-34

The proposed change revises Action Statement b as follows: "With the number of OPERABLE channels one less than required by the Minimum OPERABLE per Trip System for one or both Trip Systems, restore the inoperable channel(s) to OPERABLE status within 14 days or be in at least STARTUP within the next six hours." The Grand Gulf design includes an anticipated transient without scram (ATWS) recirculation pump trip system comprised of two systems (A and B), each receiving input from two reactor pressure vessel level channels and two reactor pressure vessel pressure channels. Any one of the four "A" input signals will initiate a recirculation pump "A" trip, any one of the four "B" input signals will initiate a recirculation pump "B" trip. Presently, the Technical Specifications require that inoperable channels be placed in the tripped condition within one hour. A channel placed in the tripped condition will result in a recirculation pump trip. To reduce the potential for any unnecessary recirculation pump trips that might be imposed on the plant by the existing Specifications, the licensee has proposed a change to the Technical Specifications that will allow for maintenance and repair. The proposed change would permit one channel in each trip system to be inoperable for a limited period of time. Based on its review, the NRC staff finds the proposed change provides a sufficiently conservative set of requirements should a trip channel become inoperable. The change provides requirements consistent with the as-built ATWS recirculation pump trip system design. Therefore, the staff finds the proposed changed acceptable.

3.3.4.2 End-of-Cycle Recirculation Pump Trip System Instrumentation (TSPS 047)

Surveillance Requirement 4.3.4.2.3, page 3/4 3-39

The proposed change revises the end-of-cycle recirculation pump trip (EOC-RPT) system response time Surveillance Requirements. The existing Surveillance Requirements specify that the four turbine control valve closure input signals to the EOC-RPT be tested every 36 months and the four turbine stop valve closure input signals to the EOC-RPT be tested every 36 months. Testing is staggered at 18-month intervals so that during one 18-month surveillance all turbine stop valve inputs are tested and during the alternate 18-month surveillance the turbine control valve inputs are tested. The proposed Surveillance Requirements specify that each 18-month test include two turbine control valve channels from one trip system and two turbine stop valve channels from the other trip system such that all channels are tested at least once every 36 months. The EOC-RPT design at Grand Gulf includes two trip systems, with each trip system receiving inputs from two turbine control valve closure sensors and two turbine stop valve closure sensors. Logic is arranged such that a trip signal will be initiated if either both turbine stop valve closure or both turbine control valve closure signals are present within either trip system. Although the frequency for the complete testing of all inputs remains at 36-month intervals, the proposed change requires testing of trip functions from both trip systems every 18 months and testing of redundant trip functions every 18 months. Based on its review, the NRC staff finds that the change provides a conservative set of plant-specific Surveillance Requirements, consistent with regulatory guidelines. Therefore, the staff finds the proposed change acceptable.

3.3.5 Reactor Core Isolation Cooling System Actuation Instrumentation (TSPS 114, 360)

(1) Limiting Condition for Operation, pages 3/4 3-45, 3/4 3-46, 3/4 3-47

The proposed change deletes the phrase "per trip system" from the heading of the "minimum operable channels" of Table 3.3.5-1. The proposed change deletes the phrase "per trip system" in the Action Statements of Table 3.3.5.1., and deletes footnotes (b), (c), and (d). The proposed change more clearly reflects the system design. The Grand Gulf design includes a single trip system for reactor core isolation cooling (RCIC) system initiation. The proposed change deletes any phrases or notes that imply a multiple trip system design. Based on its review, the staff finds that the proposed change provides a clarification in terminology without changing the requirements of the Technical Specifications. Therefore, the staff finds the proposed change acceptable.

(2) Instrumentation Setpoints, page 3/4 3-47

The proposed revision also changes the allowable value for the suppression pool water level-high trip that initiates the RCIC from "≤ 6.5 inches" to "≤ 7.0 inches." This change was made to make the value consistent with suppression pool volume. For reasons provided in paragraph (2) under Technical Specification 3.3.3 of this report, the staff concludes that the change is acceptable.

3.3.6 <u>Control Rod Block Instrumentation</u> (TSPS 009, 011, 118, 197, 237, 334, 354, 356)

Limiting Condition for Operation, pages 3/4 3-50, 3/4 3-51
Instrumentation Setpoints, page 3/4 3-52
Surveillance Requirements, pages 3/4 3-53, 3/4 3-54
Bases, pages B3/4 3-3, B3/4 3-5

This proposed change revises Tables 3.3.6-1, 3.3.6-2, and 4.3.6-1 to refer to the high power setpoint rather than the intermediate rod withdrawal limiter setpoint. For the Grand Gulf design, the high power setpoint determines when the rod withdrawal limiter function is enforcing. The Grand Gulf design does not include an intermediate rod withdrawal limiter between the low and high power setpoints. Based on its review, the NRC staff finds the proposed change corrects an error in the Technical Specifications to provide consistency with the RPCS design. Therefore, the staff finds the proposed change acceptable.

The proposed change also revises the source range monitors minimum number of operable channels requirements of Table 3.3.6-1 from four to two in Operational Condition 5, and revises the source range monitor minimum number of operable channels requirements of Technical Specification 3.3.7.6 from three to four in Operational Condition 2 and two in Operational Condition 3 or 4. In addition, the proposed change revises the Bases section for Technical Specifications 3/4 3.6 and 3/4 3.7.6 to provide additional information regarding the source range monitor's functional requirements. The purpose of the proposed change is to resolve the inconsistencies among the Technical Specifications requirements and Bases section regarding the number of source range monitors required for various Operational Conditions. Technical Specification 3.9.2 requires that two source range monitors be operable during Operational Condition 5. Two source range monitors (one in the quadrant where the core is being altered and one in an adjacent quadrant) ensure that redundant monitoring capability will be available to detect changes in the reactivity condition of the core during refueling. The operability of source range monitor input to the control rod block instrumentation in Operational Condition 5 provides diversity of rod block protection to the one-rod-out interlock and the administrative controls in place during refueling operations. In Operational Condition 3 or 4 the reactor core is homogeneous. During these conditions, the requirement for two operable source range monitors ensures redundant monitoring capability. Based on its review, the NRC staff finds that the proposed change provides consistency throughout the Technical Specifications regarding the operability requirements for source range monitors. A sufficient number of monitors are required to be operable in Operational Conditions 2, 3, 4, and 5 to provide diversity in protection and monitoring capability. Therefore, the staff finds the proposed change acceptable.

The proposed change also deletes the reference on Table 3.3.6-1 to footnote (d) on control rod block trip function 4.a, "Detector not full in." The proposed change deletes the reference to footnote (d) on trip function 4.a, which states that the intermediate range monitor (IRM) "Detector not full in" rod block function is bypassed when the IRM channels are on Range 8 or higher. From a review of the design, the licensee has determined that neither the neutron monitoring system nor the control rod block function include this feature. Based on its review, the NRC staff finds that the proposed change corrects an error in the existing Technical Specifications and provides consistency between the Technical Specifications and the design. This trip function is required to be operable in Operational Conditions 2 and 5. Removal of this automatic bypass requirement does not alter these requirements. Therefore, the staff finds the proposed change acceptable.

The proposed change also adds the reactor mode switch shutdown position to the control rod block instrumentation operability and surveillance requirements of Tables 3.3.6-1, 3.3.6-2, and 4.3.6-1. The Grand Gulf design includes circuits and components that initiate a reactor scram when the mode switch is placed in the Shutdown position. Two trip systems are provided, each including inputs from two channels associated with the mode switch. Presently, the Technical Specifications require that one channel per trip system be operable. The proposed change would require both channels in each trip system be operable. In addition, the proposed change includes Limiting Conditions for Operation and Surveillance Requirements on reactor mode switch inputs to the control rod block instrumentation. Based on its review, the NRC staff finds that the proposed changes enhance safety by correcting omissions in the existing Technical Specifications. The changes provided Technical Specifications consistent with the Grand Gulf design. Therefore, the staff finds the proposed changes acceptable.

Another proposed change revises the average power range monitor (APRM) downscale trip setpoint on Table 3.3.6-2 from $\geq$ 5% of rated thermal power to $\geq$ 4% of rated thermal power. The licensee has stated that the NSSS vendor's (GE) design specifications for the APRM downscale control rod block function delineate 4% of rated thermal power as the setpoint. Accordingly, the licensee has proposed a change to correct this error. Based on its review, the NRC staff finds that the change corrects an error in the existing Technical Specifications. The APRM downscale trip setpoint is not accounted for in the accident or transient analysis for Grand Gulf. Therefore, the staff finds the proposed change acceptable.

The algorithm for the flow biased neutron flux-upscale rod block in Table 3.3.6-2 is revised to include the T factor (= ratio of fraction of rated thermal power to the maximum fraction of limiting power density). This change was made to make this algorithm consistent with that in Technical Specification 3.2.2 and is acceptable.

Another revision was proposed to change the operability requirements and surveillance requirements for the source range monitors. The changes would be to (1) add a footnote (e) to Table 3.3.6-1 to state that the provisions of Technical Specification 3.0.4 are not applicable for entering Operational Condition 5, and (2) add a footnote (**) to Table 4.3.6-1 to require a verification that the source range monitor (SRM) detectors are "full-in" at least once every 24 hours when the detector drive motor modules are removed. When entering Operational Condition 5, Technical Specification 3.3.6 requires that the "SRM -

Detector Not Full In" input to the control rod block instrumentation be operable. To provide the operational flexibility needed to perform maintenance on the SRM detector drive motor modules in Operational Condition 5, the licensee has proposed a change to the Technical Specifications. The Technical Specification change would allow entry into Operational Condition 5 with the "SRM - Detector Not Full In" input to the control rod block instrumentation inoperable, provided a rod block is sealed in. This seal-in is accomplished by limit switches mounted on the SRM detector drive motor modules. When a motor module is removed, the limit switches actuate relays that produce the rod block. The licensee's proposal includes a surveillance requirement to verify the SRM detectors are "full-in" at least once every 24 hours with the SRM detector drive motor module removed. This would be accomplished by having a plant staff member visually inspect the SRM drive cable takeup reel at the rod gallery. Based on the review of the system design and the proposed change and following discussions with the licensee, the NRC staff does not believe that verification of the SRM detectors positions for the rod block function is necessary when the drive motor modules are removed. With the SRM detector drive motor module removed, a rod block is initiated. Once a rod block is initiated, the positions of the SRMs providing rod block input signals are irrelevant. Other Technical Specifications (3/4.9.2) address the monitoring function of the SRM during Operational Condition 5 and require periodic detector position verification. Based on its review, the staff finds that the safety function of the interlock is accomplished when the hardware is removed because the rod block is generated by the absence of the hardware. Therefore, the staff finds that entry into and plant operation in Operational Condition 5 with the interlock hardware removed is acceptable. Accordingly, except for the proposed additional Surveillance Requirement, as discussed above, the NRC staff finds that the change is acceptable.

Additional changes were proposed for the control rod block instrumentation surveillance requirements.

- One proposed change deletes footnote (e) from Table 4.3.6.1. Footnote (e) addresses surveillance requirements applicable to a reactor manual control multiplexing input From a review of the design the licensee has determined that the Grand Gulf does not include the features addressed by this specification. Based on its review, the NRC staff finds that the proposed change corrects an error in the present Technical Specifications. Therefore, the staff finds the proposed change acceptable.

- Another proposed change deletes the reference to footnote (b) from the average power range monitor (APRM), source range monitor, intermediate range monitor and recirculation flow inputs (items 2, 3, 4, and 6) to the control rod block instrument. Footnote (b) requires surveillance testing within 24 hours before startup, if not performed in the previous 7 days. The licensee has proposed deletion of this requirement since other Technical Specifications require these same features to be tested every 7 days. Based on its review, the NRC staff finds that the proposed change provides a clarification of the requirements of the Technical Specifications. Therefore, the staff finds the proposed change acceptable.

- Another proposed change deletes the functional tests required before startup from the APRM (items 2.a and 2.c) and recirculation flow (item 6.a) inputs to the control rod block instrumentation. These inputs are not required to be operable during startup and will be tested before they are required to be operable (Operational Condition 1), if they have not been tested within the previous 7 days. Based on its review, the NRC staff finds that the proposed change provides a clarification of the requirements of the Technical Specifications. Equipment is not required to be surveillance tested during conditions when the equipment is not required to be operable. Therefore, the staff finds the proposed change acceptable.

- The proposed change rewords footnote (c) to clarify the requirement to perform the surveillance test within 24 hours before control rod movement. Based on its review, the NRC staff finds that the proposed change provides a clarification of the requirement and is, therefore, acceptable.

3.3.7.1 Radiation Monitoring Instrumentation (TSPS 119, 120, 349)

Limiting Condition for Operation, pages 3/4 3-56, 3/4 3-57, 3/4 3-58
Surveillance Requirements, page 3/4 3-59

The licensee has proposed to delete offgas radiation monitors from Tables 3.3.7.1-1 and 4.3.7.1-1 along with Action Statement 71, combining them with those in Tables 3.3.7.12-1 and 4.3.7.12-1, and (2) increase the minimum number of channels operable from 1 to 2, adding applicable Action Statement 125 for the offgas post-treatment monitor in Table 3.3.7.12-1.

- The pre-treatment and post-treatment offgas radiation monitors are listed in Tables 3.3.7.1-1, 4.3.7.1-1, 3.3.7.12-1, and 4.3.7.12-1. By combining these radiation monitors in Tables 3.3.7.12-1 and 4.3.7.12-1, any unnecessary confusion and the possibility of misinterpretation of conflicting action statements will be avoided.

  The changes to Table 3.3.7.1-1 are as follows: (1) delete Items 3 and 4, the offgas pre-treatment and post-treatment radiation monitors, and (2) delete Action Statement 71 on page 3/4 3-58 because this only applies to the offgas post-treatment radiation monitor.

  The changes to Table 4.3.7.1-1 are to delete Items 3 and 4, the offgas pre-treatment and post-treatment radiation monitors. These changes have not relaxed the radiation effluent monitoring instrumentation requirements.

- The changes for the minimum number of operable channels from 1 to 2 for the offgas post-treatment radiation monitor in Table 3.3.7.12-1 are required because the monitor design consists of two detectors and both detectors must be operable to perform their intended function. This change also involves a change of presently stated Action Statement 121 to new Action Statement 125 to be consistent with two detector channels. These changes have not relaxed the radiation effluent monitoring instrumentation requirements.

Based on the foregoing evaluation, the staff concludes that the proposed changes are mostly administrative in nature and are to reflect the as-built conditions of the monitors. These changes do not relax the radiation effluent monitoring instrumentation requirements and, therefore, the staff finds the proposed changes to be acceptable.

Another change proposed for this specification is to revise the minimum channels operable column of Table 3.3.7.1-1 to require one operable channel for the dryer storage area radiation monitor (Item 10.a.3). Amendment 7 to the Grand Gulf Technical Specifications included the addition of the dryer storage area radiation monitor to Table 3.3.7.1-1. Applicable conditions for operability, alarm setpoint, measurement range and an action statement were included in Amendment 7; however, a requirement for minimum operable channels was not. Accordingly, the licensee has proposed "one" as the minimum. Based on its review, the staff finds that the proposed change is acceptable.

The proposed change to Item b. of Action Statement 75 specifies that Secondary Containment Integrity must be established with at least one standby gas treatment subsystem operating when two of the three required monitors in either the fuel handling area ventilation exhaust or the fuel handling area pool sweep exhaust are inoperable. Because a high radiation trip signal by the affected ventilation monitoring channels would isolate the auxiliary building and fuel handling area ventilation systems, the proposed change to Action Statement 75 is necessary to ensure that Secondary Containment Integrity is established. The proposed change is for the purpose of consistency with the as-built plant design and function, and it constitutes an additional requirement not contained in the current technical specification. The staff concludes that the proposed change is acceptable.

3.3.7.4 Remote Shutdown System Instrumentation and Controls (TSPS 077)

Limiting Conditions for Operation, pages 3/4 3-66, 3/4 3-67, 3/4 3-68
Surveillance Requirements, page 3/4 3-66
Based, page B 3/4 3-4

This proposed change revises the Technical Specifications to include additional Limiting Conditions for Operation and Surveillance Requirements on the remote shutdown system's switches and control circuits. Presently, the Technical Specifications only address the remote shutdown system monitoring instrumentation. The operability of the remote shutdown system instrumentation and controls ensures that the capability is available to permit shutdown of the plant from locations outside of the main control room. This capability is provided in the event control room habitability is lost. Based on its review, the staff finds the proposed changes acceptable.

3.3.7.5 Accident Monitoring Instrumentation (TSPS 367)

Limiting Condition for Operation, page 3/4 3-70

The proposed change revises the required minimum number of operable channels in Table 3.3.7.1 for the containment and drywell area radiation monitors from one each to two each. The proposed change requires that all four containment/drywell area radiation monitors be operable. If less than four channels are operable, the licensee is required to initiate the remedial actions specified. Presently, the Technical Specifications require no remedial actions until three or more channels are inoperable. Based on its review, the NRC staff finds that the proposed change provides a sufficiently conservative set of requirements should one or more instrument channels be inoperable. Therefore, the staff finds the proposed change acceptable.

3.3.7.7 Traversing Incore Probe System (TSPS 010, 050)

(1) Limiting Conditions for Operation, page 3/4-74

The proposed change revises the number of detectors required by Technical Specification for traversing incore probe (TIP) system operability from three to five. From a review of the design, the licensee has determined that five detectors are required to adequately map the core for LPRM calibration. Based on its review, the NRC staff finds that the proposed change corrects an error in the existing Technical Specifications and provides consistency between the Technical Specifications and the design. Therefore, the staff finds the proposed change acceptable.

(2) Surveillance Requirement, page 3/4-74

The proposed change revises the surveillance test required by the Technical Specifications to clarify that normalizing is required only for LPRM calibration, and is not required for individual detectors used to monitor core thermal limits. From a review of the design, the licensee has determined that a comparison of the individual detector data with the normalized data is the appropriate method for confirming operability of the TIP system when it is used for monitoring. Based on its review, the NRC staff finds that the proposed change corrects an error in the existing Technical Specifications and provides consistency between the Technical Specifications and the design. Therefore, the staff finds the proposed change acceptable.

(3) Bases, page 3/4 3-5

The proposed change revises the Bases section to provide a discussion on the TIP system surveillance tests. Based on its review, the NRC staff finds the proposed change enhances the Based section by providing additional information relevant to the Specification. Therefore, the staff finds the proposed change acceptable.

3.3.7.8 Chlorine Detection System (TSPS 346)

Limiting Conditions for Operation, page 3/4 3-75
Surveillance Requirements, page 3/4 3-75

The proposed change submits that the term "system" be changed to "channel" throughout this Technical Specification. The Grand Gulf design includes two chlorine detection channels: one that actuates control room isolation logic A, and the other that actuates control room isolation logic B. Either control room isolation logic actuation will provide control room isolation and start the respective control room emergency filtration system. Based on its review, the NRC staff finds that the proposed change is a clarification of terminology. The proposed change does not alter any of the requirements of Technical Specification. Therefore, the staff finds the proposed change acceptable.

3.3.7.9 Fire Detection Instrumentation (TSPS 073, 102, 304, 351)

Limiting Condition for Operation, pages 3/4 3-76 3/4 3-77, 3/4 3-78, 3/4 3-79, 3/4 3-80
Bases, page B 3/4 3-5

The following Technical Specification changes are proposed:

(1) Revise Table 3.3.7.9-1 format to list the fire detection instrumentation by zones and areas within each zone.

(2) Identify instrumentation by type as Function A (early warning and notification only) and Function B (actuation of fire suppression systems and early warning and notification).

(3) Clarify the associated Action Statements for Function A and Function B inoperable instrument(s).

(4) Revise the Bases to address Function A and Function B instrumentation.

(5) Incorporate additional detectors, zones, and areas into Table 3.3.7.9-1.

The proposed changes to clarify the location, type, and function of instrumentation and the changes to the Action Statements and Bases are enhancements to reflect as-built conditions. Zones, areas, and associated instrumentation are added to include those areas that contain either safety-related equipment/cables protected by fire detection instrumentation or nonsafety-related equipment/cables protected by fire detection instrumentation whose malfunction could affect fire detection instrumentation in areas containing safety-related equipment/cables. Instrumentation was also added by design changes to increase fire detection capability. These changes are enhancements to safety in that they increase the amount of fire protection equipment include in the specification, provide clarification, and are consistent with the as-built plant. The proposed changes are in accordance with the Standard Fire Protection Technical Specifications and are, therefore, acceptable.

3.3.7.10 Loose Part Monitoring System (TSPS 352)

Bases, page B 3/4 3-5

A change to the Bases for the loose-part monitoring system is proposed to distinguish between active and passive sensors. The eight active sensors provide alarm and indication functions, and the eight passive sensors at similar locations are used as replacement sensor inputs for failed sensors or to provide a change in location of the area being monitored. The passive sensors can be interchanged with the active sensors by swapping individual connectors at the loose part monitoring panel. The staff has reviewed the proposed change and found that it is consistent with the recommendations for the sensor as specified in Regulatory Guide 1.133. Therefore, the proposed change is acceptable.

3.3.7.11 Radioactive Liquid Effluent Monitoring Instrumentation (TSPS 361)

Limiting Condition for Operation, pages 3/4 3-83, 3/4 3-85

This proposed change adds a phrase "alarm and" in Item 1 of Table 3.3.7.11-1 to clarify that an alarm function exists as a specific design feature at Grand Gulf, as well as an automatic termination of liquid radwaste release. The addition of a phrase "...or Circulating Water Blowdown..." in Item 2 of Table 3/4.3.7.11-1 provides an alternate flow measurement that is more conversative than the discharge canal flow measurement, because the circulating water blowdown is only a part of the total canal discharge. This alternate method of measurement is acceptable any time the circulating water system is in service and is consistent with the as-built plant condition.

3.3.7.12 Radioactive Gaseous Effluent Monitoring Instrumentation (TSPS 120, 122, 262)

Limiting Condition for Operation, pages 3/4 3-90, 3/4 3-91
Surveillance Requirements, pages 3/4 3-92, 3/4 3-93, 3/4 3-94, 3/4 3-95

The licensee has proposed to (1) delete offgas radiation monitors from Tables 3.3.7.1-1 and 4.3.7.1-1 along with Action Statement 71, combining them with those in Tables 3.3.7.12-1 and 4.3.7.12-1, and (2) increase the minimum number of channels operable from 1 to 2, adding an applicable Action Statement 125 for the offgas post-treatment monitor in Table 3.3.7.12-1. The pre-treatment and post-treatment offgas radiation monitors are listed in Tables 3.3.7.1-1, 4.3.7.1-1, 3.3.7.12-1, and 4.3.7.12-1. By combining these radiation monitors in Tables 3.3.7.12-1 and 4.3.7.12-1, any unnecessary confusion and the possibility of misinterpretation of conflicting action statements will be avoided. The changes to Table 3.3.7.1-1 are as follows: (1) delete Items 3 and 4, the offgas pre-treatment and post-treatment radiation monitors, and (2) delete Action Statement 71 on page 3/4 3-58, because this only applies to the offgas post-treatment radiation monitor. The changes for the minimum number of operable channels from 1 to 2 for the offgas post-treament radiation monitor in Table 3.3.7.12-1 are required because the monitor design consists of two detectors and both detectors must be operable to perform their intended function. This change also involves a change of presently stated Action Statement 121 to a new Action Statement 125 to be consistent with two detector channels. These changes have not relaxed the radiation effluent monitoring instrumentation requirements. The staff concludes that the proposed changes are mostly administrative in nature and are to reflect the as-built conditions of the monitors. These changes do not relax the radiation effluent monitoring instrumentation requirements and, therefore, the proposed changes are acceptable.

Another change to this specification regards the standby gas treatment system (SGTS) exhaust radiation monitors during testing of the SGTS. The licensee originally requested this proposed change in a letter dated April 10, 1984. Subsequently the request was denied by the staff (Order by H. R. Denton, NRC, dated April 18, 1984) because the proposed applicable Action Statement 127 would allow SGTS operation without the radiation monitor in service as long as grab sample and analysis requirements were met for a 30-day period. However, the licensee resubmitted his proposed changes with additional operational information and further clarification of the monitor design. The licensee states that during the SGTS surveillance and secondary containment integrity tests, the fuel handling and auxiliary buildings ventilation systems must be isolated and the SGTS provides the sole means of ventilation for these buildings. Furthermore, the licensee also states that the SGTS radiation monitors (low range detector) in Table 3/4.3.7.12-1 are intended for use to monitor radioactivity releases during the SGTS test periods, whereas the SGTS radiation monitors (middle and high range detectors) listed in Table 3.3.7.5-1 are intended for use during accident conditions. Therefore, during the periods of SGTS testing and secondary containment integrity demonstration, the SGTS monitor listed in Table 3/4.3.7.12-1 also serves as normal fuel handling and auxiliary building ventilation exhaust monitor. We find the proposed changes acceptable since they are consistent with other building ventilation exhaust monitoring requirements, as specified in NUREG-0473.

Another proposed change addes footnote 5 to Technical Specification Table 4.3.7.12-1 and makes this footnote applicable to the Channel Functional Test requirements for the flow rate monitors in the exhaust monitor systems for the radwaste building ventilation, containment ventilation effluent, turbine building ventilation, and fuel handling area ventilation. The footnote clarifies the requirement that the measured flow rates be compared to the expected design flow rates for the existing plant conditions for these systems. The staff concludes that this change is acceptable.

<u>3.3.8 Plant Systems Actuation Instrumentation</u> (TSPS 033, 083, 345)

Limiting Condition for Operation, pages 3/4 3-98, 3/4 3-98a
Instrumentation Setpoints, page 3/4 3-99
Surveillance Requirements, page 3/4 3-100
Bases, page B 3/4 3-6

The proposed change revises Tables 3.3.8-1, 3.3.8-2, and 4.3.8.1-1 to include Limiting Conditions for Operations and Surveillance Requirements for the suppression pool makeup system actuation instrumentation, and revises the Bases section to address this instrumentation. Presently, Technical Specification 3/4 6.3.4 contains operability and surveillance requirements for the suppression pool makeup system, but does not address the associated actuation instrumentation. The proposed change adds the actuation instrument to Tables 3.3.8-1, 3.3.8-2 and 4.3.8.1-1. Based on its review, the NRC staff finds that the proposed change reflects the as-built plant design and provides a sufficiently conservative set of requirements. The required minimum number of operable channels satisfy the single-failure criterion; the applicable Operational Conditions are consistent with the Operational Conditions requirements for the suppression pool makeup system in accordance with Technical Specification 3.6.3.4; the Surveillance Requirements are similar to those provided for instruments used for other reactor protection functions, Therefore, the staff finds the proposed change acceptable.

The proposed change reduces the allowable value for the reactor vessel water level-high, feedwater system/main turbine trip system trip function, Item 2.a on Table 3.3.8-2, from $\leq$ 55.7 in. to $\leq$ 54.1 in.

The licensee has stated that the proposed change corrects an error in the Technical Specifications. The present allowable value was determined assuming wide-range instrumentation accuracy in the setpoint calculations, whereas narrow-range instrumentation provides the trip signal. The revised allowable value reflects the accuracy of the narrow-range instruments. In response to a request from the NRC staff, the licensee is participating in a BWR Owner's Group effort to provide more detailed information on their setpoint methodology. The final acceptability of the Grand Gulf setpoint methodology, trip setpoints, and allowable values will be addressed in another supplement to the SER. The staff concludes with reasonable assurance, based on staff participation in meetings with the BWR Owner's Group working group on setpoint methodology, that the forthcoming more-detailed information on setpoints and setpoint methodology being developed by this group will verify the acceptability of the proposed setpoints. In the interim, the staff finds the proposed change acceptable.

3.3.9 Turbine Overspeed Protection System (TSPS 148)

Limiting Condition for Operation, page 3/4 3-101
Surveillance Requirement, page 3/4 3-101
Bases, page B 3/4 3-6

This change adds a new specification and bases for the turbine overspeed protection system. The proposed specification meets the staff's SER requirements for testing the overspeed protection system, and is, therefore, acceptable.

3.4.1.1 Recirculation System (TSPS 041, 382)

(1) Limiting Condition for Operation, page 3/4 4-1

This proposed change revises Action Statements a and b of Technical Specification 3.4.1.1. The change is to incorporate the additional requirement of immediately reducing power to or below the 80% power specified by the 100% rod line of Figure B.3/4.2.3-1 for the conditions of one or two recirculation loops inoperable. The staff has reviewed the proposed change and found that the change is prudent and acceptably resolves the thermal-hydraulic stability concerns for Grand Gulf Unit 1 for two-loop operation. However, single-loop operation is not permitted until the licensee's request for single-loop operation and the associated Technical Specification changes addressing the thermal-hydraulic stability concerns are submitted and approved by NRC staff.

(2) Surveillance Requirement, page 3/4 4-1
Bases, page B 3/4 4-1

This proposed change adds Surveillance Requirement 4.4.1.1.1 to verify once every 24 hours that both recirculation loops are in operation while in Operational Conditions 1 and 2 and also adds a description of the operation of the recirculation loop flow control valves to the Bases. The change does not alter or delete any currently existing specifications. The proposed revision to the Bases provides additional detail regarding operation of the recirculation flow control valves. Based on its review, the staff concludes that these changes are acceptable.

3.4.1.2 Jet Pumps (TSPS 024, 236)

Surveillance Requirements, page 3/4 4-2

This proposed change revises the Surveillance Requirement by deleting the requirement that the recirculation flow control valves be in the same position when performing the surveillance. The revised requirement specifies that the surveillance is to be performed with the recirculation loop flows within the mismatch limits of Technical Specification 3.4.1.3. The existing specification requires that the Surveillance Requirement be performed with the flow control valves (FCV) in the same position. Changing the position of one FCV relative to the other has the effect of changing the flow in one recirculation loop relative to the other loop. The proposed change would allow different FCV positions, and thus different flows, provided that the flow mismatch is within the normal mismatch limits of Specification 3.4.1.3. The staff has reviewed the proposal and notes that even at the same FCV position, relative loop flows are different because of differing flow path resistances and individual pump characteristics. The effect of the proposed change on measured parameters is, therefore, expected to be insignificant. The staff concludes that the change is acceptable.

Another revision is proposed to allow present Surveillance Requirement 4.4.1.2 (renumbered to 4.4.1.2.1) to be performed "with Thermal Power in excess of 25% of Rated Thermal Power" instead of the present "before exceeding 25% of Rated Thermal Power." A new Surveillance Requirement (4.4.1.2.2) is added to provide a Technical Specification 4.0.4 exemption, provided the diffuser to lower plenum differential pressure of each individual jet pump is determined to be within 50% of the loop average within 72 hours after entering Operational Condition 2 and at least once every 24 hours thereafter. The value of 50% of the loop average is qualified, by addition of footnote *, to explain that this is an initial value and that the final value will be determined during the startup test program with any required change submitted to the Commission within 90 days of test completion. The proposed Surveillance Requirement 4.4.1.2.2 will allow entry into Operational Conditions 1 and 2 without having to perform present Surveillance Requirement 4.4.1.2; however, jet pump Operability is required to be determined within 72 hours after entering Operational Condition 2 and at least once every 24 hours thereafter by verifying that the diffuser to lower plenum differential pressure is within specified limits. Entry into Operational Condition 2 is necessary to perform the surveillance required to demonstrate jet pump operability. Operational Condition 2 operation is needed to achieve power levels sufficient for meaningful measurements of flow and differential pressure (dp). When power and flow conditions are too low, the effects of natural circulation, moderator subcooling changes, and varing core dp result in large data uncertainties. These large uncertainties also necessitate different criteria for demonstrating jet pump operability when power is less than 25% of rated thermal power (RTP). The 50% criteria (when power is less than 25% RTP) is based on an extrapolation of information provided by General Electric in Service Information Letter (SIL) Number 330, dated June 9, 1980. SIL No. 330 discusses jet pump beam cracks and jet pump displacement that have occurred at three plants. With a displaced jet pump, flow deviation from average was 70%. This deviation is determined by differential pressure measurement, which is the criterion proposed by the licensee. The licensee will also verify the 50% criteria during the startup test program when plant-specific data will be generated. Any required changes will be submitted within 90 days of test completion. Above 25% RTP jet pump operability requirements are unchanged from the existing specification. The staff has reviewed the licensee's proposed changes to Technical Specification 4.4.1.2 and finds them acceptable because they are necessary for meaningful surveillance measurements. The proposed changes also include a safety enhancement in that a requirement for jet pump operability when power level is less than 25% RTP for an extended period of time is added. The staff concludes that the proposed changes are acceptable.

<u>3.4.1.4 Reactor Coolant System - Idle Recirculation Loop Startup</u> (TSPS 123)

Limiting Condition for Operation, page 3/4 4-4

This specification prohibits startup of a recirculation loop unless the temperature differential between the reactor vessel steam space coolant and the bottom head drain line coolant is $\leq$ 100°F and certain other conditions are met. The change proposed is to add a footnote to the 100°F stating that the temperature differential limit is not applicable below 25 psig.

The temperature of the vessel steam space coolant is obtained from a pressure measurement signal that is then converted into a temperature value in a process computer by reference to a standard pressure/temperature saturation curve. The licensee also stated that by means of the available temperature indication it is not possible to measure properly the differential temperature specified in the current Technical Specification 3.4.1.4 when the reactor is at ambient pressure. A steam space coolant temperature of 212°F will be indicated when in fact the true temperature may be much lower. Thus, the present Technical Specification places unnecessary restrictions on recirculation loop operation. The proposed change is to make this Technical Specification applicable only when the vessel pressure is greater than 25 psig.

The licensee stated that a fatigue evaluation was performed for the Grand Gulf Nuclear Station Unit 1 reactor vessel and components by GE in accordance with Subsections NA, NB, and Appendix I of Section III of the ASME Boiler and Pressure Vessel Code, Summer 1976 Addenda and later editions. The analysis included a 100°F temperature change in the reactor vessel bottom head for all predicted reactor startup-shutdown and scram cycles, 20 cold liquid injections, and one 348°F temperature change during preoperational testing. The analysis predicted a cumulative fatigue usage factor of 0.5509 at the end of the unit's 40-year service life.

Furthermore, the licensee stated that from the fatigue evaluation, the liquid control nozzle was found to be the most limiting bottom head component. With the proposed Technical Specification change, the licensee has determined that the largest temperature difference that could occur in this nozzle would be 187°F at 25 psig. However, the licensee also stated that this large temperature difference (187°F) would be possible only if the following highly unlikely sequence of events were to occur:

(1) recirculation pumps were removed from service

(2) reactor vessel was held stable at 25 psig for an extended period of time

(3) the drywell environment was cooled to a low enough temperature to allow the bottom head and the moderator therein to cool to 80°F

(4) the reactor vessel dome was at saturated condition

(5) all the above unusual conditions and subsequent start of an idle recirculation pump occurred

Assuming that the preceding low probability sequence of events did occur, the licensee performed an additional fatigue usage evaluation showing that there is a sufficient margin for at least 130 events of the 187°F temperature difference to occur during the service life of the vessel without exceeding a fatigue usage factor of 1.0. Based on a review of the information submitted by the licensee, the staff believes that there is a low occurrence rate (not to exceed 10 events during the unit's 40-year service life) that this series of events would occur and thus concludes that an adequate margin to allowable fatigue usage factor of 1.0 exists for temperature difference larger than 100°F. The proposed change to Technical Specification 3.4.1.4 does not reduce safety margins in the plant design and will not increase the probability or consequence of either a new or a previously analyzed accident. Therefore, the proposed change of Technical Specification is acceptable.

3.4.2 Reactor Coolant System Safety/Relief Valves (TSPS 023, 257)

Limiting Condition for Operation, pages 3/4 4-5, 3/4 4-6

The proposed change revises Technical Specifications 3.4.2.1 and 3.4.2.2 to provide explicit remedial actions to be taken in the event that either the pressure actuation trip system or low-low set function of a safety/relief valve is inoperable. The Grand Gulf design includes redundant trip logics for the pressure actuation trip system and low-low set pressure relief function. Presently, the Technical Specifications provide no specific requirements with regard to the inoperability of the two trip systems. One possible interpretation would allow one of the two trip system logics to be inoperable indefinitely; another would require plant shutdown within 12 hours if one or both logics were inoperable. To provide a set of specific requirements should one or both logic systems be inoperable, the licensee has proposed additional Action Statements for Technical Specifications 3.4.2.1 and 3.4.2.2. Based on its review, the NRC staff finds the proposed Action Statements provide a sufficiently conservative set of requirements should one or both channels be inoperable. Therefore, the staff finds the proposed change acceptable.

Another proposed change revises the Limiting Condition for Operation for Technical Specification 3/4 4.2.1 to specifically address the safety/relief valve tailpipe pressure switches. Presently, Action Statement c to Technical Specification 3.4.2.1 provides remedial actions to be taken if the safety/relief pressure switches are inoperable. The proposed change adds a specific reference to these switches in the Limiting Conditions for Operation. Based on its review, the NRC staff finds the proposed changes are administrative in nature and do not alter the requirements of the existing Technical Specifications. Therefore, the staff finds the proposed change acceptable.

3.4.3.2 Reactor Coolant System Operational Leakage (TSPS 028, 032)

(1) Limiting Condition for Operation, page 3/4 4-8
Surveillance Requirements, pages 3/4 4-9, 3/4 4-10
Bases, page B 3/4 4-2

The proposed change is to revise Action Statement d of the Technical Specification and Surveillance Requirement 4.4.3.2.3 to reflect that high/low pressure interface valve interlocks as well as alarms are present. It is further proposed to delete the current Table 3.4.3.2-2 and insert new Tables 3.4.3.2-2 and 3.4.3.2-3 to incorporate setpoints for the interlocks and for alarms not presently listed. The term "Leakage" is deleted from the description of the monitors in Action Statement d and in Table 3.4.3.2-2, entitled "Reactor Coolant System Interface Valves Pressure Leakage Monitors." Action Statement d is further revised to delete the requirement to restore Operability of the inoperable monitors within 7 days or verify the pressure to be less than the alarm setpoint at least once every 12 hours. It is also proposed to revise Bases B 3/4.4.3.1 to reflect that the RCS leakage detection system also measures leakage from fluid systems in the drywell.

The changes to Action Statement d and Surveillance Requirement 4.4.3.2.3 to specify interlocks and alarms is an enhancement, which clarifies the function of the equipment being tested.

Replacing current Table 3.4.3.2-2 with new Tables 3.4.3.2-2 and 3.4.3.2-3 results in additional surveillance requirements. The additional surveillance results from additional equipment being listed in the new tables relative to the current Table. All the equipment in the current table is present in the new tables. In addition, the tables specify the "alarm" and "interlock" function of the instrumentation separately. The staff has reviewed the alarm and interlock setpoints and concludes that, in each case, the setpoint is below the design pressure of the low-pressure system involved.

Deleting the term "Leakage" from the description of the monitors and in Action Statement d is acceptable because the monitors do not measure leakage rate but are primarily intended to provide an operator alarm or interlock for overpressure protection.

Deleting the requirement to "restore operability of the inoperable monitor within 7 days or verify the pressure to be less than the alarm point at least once per 12 hours" makes the specification consistent with Limiting Condition for Operation 3.4.3.2, which allows 1 gpm leakage from any reactor coolant system pressure isolation valve. This is seen as follows: Any leakage past an inboard pressure isolation valve will pressurize the space between the inboard and outboard pressure isolation valves. Limiting Condition for Operation 3.4.3.2 allows some small leakage and thus implies that the intervalve space may be normally pressurized. As a consequence, verification that the pressure is below the alarm or interlock setpoint may not be possible, although the valve leakage specification is satisfied. The proposed specification requires that the reactor be shut down within 12 hours if a monitor or interlock cannot be restored to operable within 30 days.

The staff concludes that the proposed changes are acceptable.

(2) Table 3.4.3.2-1, Reactor Coolant System Pressure Isolation Valves, page 3/4 4-10

The staff has reviewed the proposed list of reactor coolant system pressure isolation valves (Table 3.4.3.2-1) and found that the list includes the valves required by the staff in addition to other valves appearing in the table. The valves required by the staff to be leak rate tested are listed below.

Valve Number	System
E21-F005	LPCS
E21-F006	
E22-F004	HPCS
E22-F005	
E12-F008	RHR
E12-F009	
E12-F041 A, B, C	
E12-F042 A, B, C	
E51-F065	RCIC
E51-F066	

The table as proposed meets the staff's minimum requirements and is therefore acceptable.

3.4.4 Chemistry (TSPS 055)

Surveillance Requirement, page 3/4 4-12

Technical Specification Surveillance Requirement 4.4.4.c is presently worded "Continuously recording the conductivity of the reactor coolant, or, when the continuous recording conductivity monitor is inoperable for up to 31 days, obtaining an in-line conductivity measurement at least once per:" The proposed change is to delete the phrase "for up to 31 days."

The proposed change in deleting the phrase "for up to 31 days" will make the revised Surveillance Requirement 4.4.4.c consistent with regulatory guidelines and is, therefore, acceptable.

3.4.6.1 Reactor Coolant System Pressure/Temperature Limits (TSPS 160, 219)

Limiting Conditions for Operation, pages 3/4 3-17, 3/4 3-19
Surveillance Requirements, page 3/4 3-18
Bases, pages 3/4 B 2-5, 3/4 B 4-4, 3/4 B 4-5

The proposed changes will correct typographical errors, clarify technical specifications and their bases, render the Technical Specification consistent with the as-built plant, and update the pressure-temperature limits to comply with the closure-flange safety margins in Paragraph IV.A.2 Appendix G, 10 CFR 50, that became effective on July 26, 1983. The staff therefore concludes that the proposed changes are acceptable.

3.4.7 Main Steam Line Isolation Valves (TSPS 243)

Surveillance Requirement, page 3/4 4-22

The proposed change addes a footnote to Technical Specification 4.4.7 to specify differences in testing assumptions for the minimum and maximum main steam line isolation valve (MSIV) closing time surveillance. The minimum closure time is measured from the start of valve motion. The maximum closure time is measured from initiation of the actuation signal. The minimum and maximum MSIV closure times are based upon different considerations. The minimum closure time is used in overpressurization analyses and transient analyses where a faster closure time leads to a more severe event. In these analyses, the initiating event is the actual valve closure with no actuation signal included. The proposed footnote for minimum closure time does not include an actuation signal and is, therefore, consistent with the safety analyses.

Maximum valve closure time is considered for the purpose of minimizing the release of radioactive material. By definition, a maximum time must include all contributors from the initiating signal until full valve closure, including the actuation signal response time. The footnote proposed by the licensee clarifies the requirement that actuation signal response time be included in the measured maximum closure time. The changes proposed by the licensee clarify the existing specification and are therefore acceptable.

3.4.9 Residual Heat Removal System (TSPS 272)

Limiting Condition for Operation, page 3/4 4-24
Surveillance Requirement, page 3/4 4-24

The proposed change incorporates into Action Statement 2 of Technical Specification 3.4.9.1 and into Surveillance Requirement 4.4.9.1 the provision for substituting one operating recirculation pump for one operating shutdown cooling mode loop of RHR as is allowed by the subject Limiting Condition for Operation. These revisions do not change any setpoints, procedures, Limiting Conditions for Operation, or Surveillance Requirements. The changes are administrative and serve only to clarify Action Statement 2 with regard to one recirculation pump being an acceptable alternative to having one RHR shutdown cooling mode in operation for satisfying Limiting Condition for Operation 3.4.9.1. The changes are, therefore, acceptable.

3.5 Emergency Core Cooling Systems (TSPS 126, 162, 168, 256, 309, 310)

(1) Action Statement for RHR Systems, page 3/4 5-1

This proposed change adds footnote ** to Action Statement a.4 of Technical Specification 3.5.1. The change clarifies the Action to be taken whenever two or more RHR subsystems are inoperable. This footnote is identical to footnote *, which is applicable to Action Statements b.3 and d.3 of Technical Specification 3.5.1. This change is purely administrative and is made to achieve consistency among Action Statements 3.5.1.a.4, 3.5.1.b.3, and 3.5.1.d.3. Therefore, the change is acceptable.

(2) Surveillance Requirement for the LPCI Systems, page 3/4 5-4

A revision is proposed to change the high-pressure alarm setpoint of the LPCS system from 580 +20, -0 psig to $\leq$ 600 psig. This proposed revision also changes the high-pressure alarm setpoint of the LPCI subsystems from 480 +20, -0 psig to $\leq$ 493 psig and clarifies Surveillance Requirement 4.5.1.c.2.a to indicate that both high- and low-pressure alarms exist for the subject systems. The purpose of these alarms is to advise the operator of leakage past the motor-operated LPCS or LPCI injection valves. This leakage could lead to overpressurization of the low-pressure LPCS or LPCI systems. The specification changes proposed by the licensee eliminate the lower bound on the setpoints. This is conservative because a lower setpoint will provide earlier indication of leakage. The proposed upper limit of 600 psig for the LPCS alarm setpoint and 493 psig for the LPCI alarm setpoint have been discussed with the licensee in a telephone conversation on June 29, 1984. The staff indicated that lower values for the upper limits are desirable because system design pressures are 600 psig and 500 psig for the LPCS and LPCI systems, respectively. The licensee and staff agreed that upper limits of 575 psig and 475 psig for LPCS and LPCI, respectively, provided additional margin and were acceptable.

The change to the wording of Surveillance Requirement 4.5.1.c.2.a is strictly a clarification to indicate the difference between the discharge line high-pressure alarm and the "keep filled" system low-pressure alarm.

The staff has reviewed the licensee's proposed changes and concludes that provided the upper limit setpoint for the LPCS system high-pressure alarm does not exceed 575 psig, and the upper limit setpoint for the LPCI system high-pressure alarm does not exceed 475 psig, the changes are acceptable.

(3) Bases for High-Pressure Core Spray System, pages B 3/4 5-1, B 3/4 5-2

A revision to the Bases for the ECCS Technical Specifications is proposed to achieve consistency among the FSAR, design documentation, and the Technical Specifications. The stated system operating differential pressure range and HPCS pump capacity values are revised to be consistent with the information provided in FSAR Chapter 6.3.1, design documents, and Technical Specifications 3/4.5.1 and 3/4.5.2. Minor editorial corrections are also proposed to improve the readability of the Bases section. The staff concludes that these changes are acceptable.

(4) Suppression Pool Operability for ECCS Systems

Limiting Condition for Operation, pages 3/4 5-8, 3/4 5-9
Surveillance Requirement, page 3/4 5-9

The change would revise the minimum water level in the suppression pool during Operational Conditions 1, 2, and 3 from 18 ft. 4 3/4 in. to 18 ft. 4½ in. This change is acceptable for reasons given in the evaluation of Technical Specification 3.6.3 of this report.

Another proposed change would revise the minimum allowable water level in the suppression pool during Operational Conditions 4 and 5 from 12 ft. 5 in. to

12 ft 8 in. The proposed change increases the minimum water level required during Operational Conditions 4 and 5 and is, therefore, more conservative than the existing specification. Supporting calculations consistent with Regulatory Guide 1.1 demonstrate that sufficient net positive suction head (NPSH) is available to the ECCS pumps at the proposed minimum level. The staff finds that the proposed change is acceptable.

3.6.1.2 Containment Leakage (TSPS 057, 067, 294)

Limiting Condition for Operation, pages 3/4 6-2, 3/4 6-3
Surveillance Requirements, pages 3/4 6-3, 3/4 6-4

The proposed changes clarify the leakage requirements for isolation valves and penetrations. The terms "ECCS and RCIC" would be deleted from the appropriate sections of the text. In addition, an editorial change to Technical Specification 3.6.1.2.b and Action Statement b consists of removing a portion of the text and placing it in a footnote. As presently written, this specification requires consideration of only ECCS and RCIC containment isolation valves. Other isolation valves should also be included in the group of valves that are subject to containment leak rate requirements. Thus, deleting the term "ECCS and RCIC" from the appropriate sections of the text will broaden the scope of the Technical Specification section. Also, the additional Technical Specification change in this section is editorial in nature. Therefore, the proposed changes are acceptable.

In other changes Surveillance Requirement 4.6.1.2.c.1 would be revised to provide clarification concerning the method to be used for verification of the accuracy of Type A containment leakage rate testing (Appendix J, 10 CFR 50). In addition, Surveillance Requirement 4.6.1.2.c.3 would be revised to indicate that the required quantity of gas injected into, or bled from, the containment during the supplementaal test must be between 0.75 L_a and 1.25 L_a. These changes are considered to be clarifications and are in compliance with Appendix J of 10 CFR 50. The staff therefore finds the proposed Technical Specification changes acceptable.

In another change, Surveillance Requirement 4.6.1.2.k would be revised to clarify that the 25% surveillance interval extension permitted by Technical Specification 4.0.2 is not applicable to those time periods specified by Appendix J of 10 CFR 50. The NRC staff has reviewed the proposed change and has determined that it improves safety in that it represents a clarification of the Technical Specifications for the purpose of helping ensure compliance with of Appendix J of 10 CFR 50. The staff therefore concludes that the proposed action is acceptable on the basis of technical and safety considerations.

3.6.1.3 Containment Air Locks (TSPS 235)

Surveillance Requirement, page 3/4 6-6

For this change, Surveillance Requirement 4.6.1.3.a would be revised to add a footnote to both "72 hours" time limits, which will specify that the provisions of Technical Specification 4.0.2 are not applicable. This change indicates that no extension of the 72-hour time limit to demonstrate operability of each containment air lock is allowed and ensures that the Technical Specifications are in compliance with Appendix J of 10 CFR 50. Therefore, the staff finds the proposed change to be acceptable.

3.6.1.4 Main Steam Isolation Valve (MSIV) Leakage Control System (LCS) (TSPS 195, 229)

Surveillance Requirements, page 3/4 6-7

One of the proposed changes in this specification concerns changes to the Surveillance Requirements of the main steam isolation valve leakage control system (MSIV-LCS) Technical Specification 4.6.1.4 that require that Operability of the heaters for each MSIV-LCS subsystem be demonstrated by verifying that the heaters draw 7.8 to 9.5 amperes per phase. However, because only the inboard MSIV-LCS has heaters, a change from "heater" to "inboard heater" has been proposed. Also, the amperage range for demonstrating heaters Operability has been added to the 31-day Surveillance Requirement and reworded to read "8.65 amperes + 10%" instead of "7.8 to 9.5 amperes." These proposed changes make the Technical Specification consistent with the as-built plant as specified in FSAR Section 6.7.1 and add Surveillance Requirements not presently in the Technical Specifications. Therefore, the staff concludes that the proposed changes are acceptable.

Another proposed change concerns clarification of the MSIV-LCS Technical Specification Surveillance Requirement 4.6.1.4.c.2, and the addition of a note to clarify the valve lineup during the blower surveillance. The new pressure and flow values were determined during preoperational and startup testing using the specified valve lineup. These changes are consistent with the safety analysis and are therefore acceptable.

3.6.2.3 Drywell Air Locks (TSPS 031)

Surveillance Requirement, page 3/4 6-16

Surveillance Requirements 4.6.2.3.a and 4.6.2.3.b are changed to be consistent with Surveillance Requirements 4.6.1.3.a and 4.6.1.3.b. The proposed change makes the Technical Specification consistent with Appendix J of 10 CFR 50. Therefore, the staff finds the proposed changes acceptable.

3.6.2.5 Drywell Internal Pressure (TSPS 127)

Limiting Condition for Operation, page 3/4 6-18
Bases, page B 3/4 6-3

The proposed change is to correct the permissible lower limit of the drywell to containment differential pressure from -0.1 to -0.26 psid and the corresponding Bases. An additional change is proposed for the Bases, which states that the 2.0-psid limit for positive drywell to containment pressure will "not allow clearing of the top vent." The change in the minimum drywell-to-containment differential pressure was previously submitted by letter from the licensee dated September 9, 1983. The change makes the Technical Specification representative of the plant design that incorporates changes to resolve one of the Mark III containment issues (Humphrey concerns). The proposed changes are needed to provide consistency with the safety analyses. Therefore, the staff finds the Technical Specification changes to be acceptable.

3.6.3.1 Suppression Pool (TSPS 126, 168)

Limiting Condition for Operation, page 3/4 6-20
Surveillance Requirements, page 3/4 6-21
Bases, page B 3/4 6-4

One of the proposed changes is to correct the suppression pool high water volume from 138,851 to 138,701 cubic feet to be consistent with the FSAR. Accordingly, the Bases section on suppression pool volume was changed to be consistent with the Technical Specification. Therefore, the staff finds the proposed change to be acceptable.

Additional proposed changes to the subject Technical Specifications are:

(1) Regarding Technical Specifications 3.5.3 and 3.6.3.1, revise the suppression pool low and high water levels (depths) to 18 ft 4-1/12 in. and 18 ft 9-3/4 in., respectively, to make them consistent with the FSAR.

(2) Delete the reference to Operational Conditions 1 or 2 in Limiting Condition for Operation 3.6.3.1.b, Action Statement 3.6.3.1.b, and Surveillance Requirement 4.6.3.1.b.

(3) Expand Bases 3/4.6.3, "DEPRESSURIZATION SYSTEMS," to provide bases for:

    (a) suppression pool volumes
    (b) suppression pool levels (depths)
    (c) suppression pool temperatures

The changes to the suppression pool water levels are made to be consistent with the volumes used for the safety analyses. The deletions of the reference to Operational Condition 1 or 2 are considered enhancements that make the Limiting Conditions for Operation, Action Statements, and Surveillance Requirements consistent with the Applicability Statement. The changes to Bases 3/4.6.3 provide substantial clarification and are consistent with the safety analyses. The staff finds the proposed Technical Specification changes to be acceptable.

3.6.3.2 Containment Spray (TSPS 012, 169, 233)

3.6.3.3 Suppression Pool Cooling

Limiting Conditions for Operation, pages 3/4 6-24, 3/4 6-25
Surveillance Requirement, page 3/4 6-24
Bases, page B 3/4 6-4

The proposed changes follow:

(1) change "SSW heat exchanger" to "RHR heat exchanger" in Technical Specification 3.6.3.2.b and 3.6.3.3.b

(2) add the containment spray spargers to Technical Specification 3.6.3.2.b

(3) revise Action Statement b of Technical Specification 3.6.3.2 to be consistent with Action Statement b of Technical Specification 3.6.3.3

(4) change the time in Action Statement a of Technical Specification 3.6.3.3 from 7 days to 72 hours

(5) add new Surveillance Requirement 4.6.3.2.d

(6) revise Bases 3/4.6.3 to reflect the added Surveillance Requirement

The change from standby service water (SSW) to reactor heat removal (RHR) heat exchanger corrects an error in terminology and is made to reflect correct nomenclature. The addition of the containment spray spargers to Technical Specification 3.6.3.2.b reflects the system design and is proposed to ensure system operability. The changes to the Action Statements are enhancements to achieve consistency between the containment spray and suppression pool cooling modes of RHR operation. The addition of the requirement to perform an air or smoke flow test to Surveillance Requirement 4.6.3.2 and the revision to Bases 3/4.6.3 constitutes an additional requirement not presently included in the Technical Specifications. The licensee contends that the design of the containment spray system is such that nozzle obstruction should not occur unless caused by maintenance activities; therefore, the surveillance frequency should not be time dependent but instead should be coordinated with the completion of applicable maintenance activities. The containment spray nozzles were initially air-tested during the preoperational test phase and no maintenance has been performed on the system since that time that could cause nozzle blockage. Based on the above discussion, the staff finds the proposed Technical Specification changes to be acceptable and necessary.

An additional change to the Bases is to add a statement confirming that Surveillance Requirements 4.5.1.b and 4.6.3.2.b provide adequate assurance that the containment spray system will be operable when required. Sufficient flow through the RHR heat exchangers will ensure sufficient flow to the containment spray nozzles because the minimum acceptable flow and total developed head values, stated in the surveillance requirement, account for inherent system losses. This change is an enhancement that clarifies the Bases for the containment spray system specification. Therefore, the staff finds the proposed Technical Specification change to be acceptable.

<u>3.6.3.4 Suppression Pool Makeup System</u> (TSPS 004, 312)

Surveillance Requirements, page 3/4 6-26
Bases, page B 3/4 6-5

One proposed change to Surveillance Requirements is to require that both refueling gates be in the stored position or be otherwise removed from the upper containment pool to provide assurance that an adequate source of water exists for the suppression pool makeup system. The staff concludes that this change provides additional assurance that an adequate source of water exists for the suppression pool makeup system, and is, therefore, acceptable.

Another proposed change revises Technical Specification 4.6.3.4.a.1 to require that the level be measured from above the pool bottom in the dryer/separator storage area. In addition, the proposed change modifies the associated Bases section for Technical Specification 3/4.6.3 to discuss the consequences of inadvertent draindown of the suppression pool makeup system. The proposed change is provided as a clarification that the reference point for water level in the upper containment pool is the dryer/separator storage area of the pool floor. The licensee has confirmed that with this clarification the quantity of water provided is sufficient to account for post-accident entrapment volumes, ensuring the long-term energy sink capabilities of the suppression pool and maintaining the water coverage over the uppermost drywell vents. Based on its review, the NRC staff finds that the proposed change provides a clarification of the Surveillance Requirements that does not change the specified level or the bases for the specified level. Therefore, the staff finds the proposed change acceptable. The proposed change to revise the Bases section of the Technical Specifications provides a discussion on the consequences of and the safeguards used to prevent opening of the suppression pool makeup dump valves. Based on its review, the NRC staff finds the proposed change enhances the Bases section by providing additional information relevant to the Specification. Therefore, the staff finds the proposed change acceptable.

3.6.4 Containment and Drywell Isolation Valves (TSPS 020, 306)
3.6.6.2 Secondary Containment Automatic Isolation Dampers/Valves

Limiting Condition for Operation Surveillance Requirements, pages 3/4 6-27 through 3/4 6-40, 3/4 6-42, 3/4 6-44, 3/4 6-48 through 3/4 6-52.

The proposed changes are to revise the maximim isolation times of a number of valves in Table 3.6.4-1. Additional changes would revise an incorrect penetration number, move valves E12-F042A and E12-F042B from Section 1.a to Section 2.a, and add information designating the divisional power supply associated with the valves in Table 3.6.4-1 and 3.6.6.2-1. All isolation valves having analytical closure times in Table 3.6.4-1 will be changed to reflect those closure times. The remaining revisions to the maximum isolation times are for valves that have no analytical closure times. The original closure times of the listed valves were based mainly on design or purchase specifications values. The proposed revisions to the valve closure times are based on previous test data with an appropriate margin. The value of the margin was obtained from methodology contained within ASME Section XI. The proposed changes provide a consistent basis for the maximum isolation times and provide a realistic measure of valve performance. The remaining changes are administrative to provide additional improvements. The proposed changes to Tables 3.6.4-1 and 3.6.6.2-1 do not affect the capability of the primary and secondary containments to perform their safety function. Therefore, the staff concludes that the proposed changes are acceptable.

In order to conform to "Type C" (Appendix J of 10 CFR 50) leak testing requirements, the licensee has committed by letter dated September 12, 1983, to pneumatically test selected valves that had previously required hydrostatic testing. Revisions to Table 3.6.4-1 are proposed to require pneumatic testing of those selected valves by deleting various footnote notations. Also, regarding valves E51-F251 and E51-F252, footnote (e) notation would be changed to (c) notation. The proposed changes are consistent with the licensee's commitment. It should be noted that penetration 27 has previously been identified by the licensee as requiring pneumatic testing of its isolation valves. Subsequent investigation by the licensee has determined that the penetration is below the minimum suppression pool drawdown level; therefore, exposure to containment atmosphere will not occur and pneumatic testing of its valves is not required. In addition, the licensee indicated a recent design change has made it necessary to change the type of hydrostatic test (distinguished by the change from footnote (e) to (c) notation) that should be performed on valves E51-F251 and E51-F252. Thus, the footnote notation was modified appropriately. Based on its review, the staff finds the proposed changes to be acceptable.

3.6.7.2 Containment and Drywell Hydrogen Ignition System (TSPS 069)

Limiting Condition for Operation, page 3/4 6-57
Surveillance Requirement, page 3/4 6-57

The proposed change replaces present Technical Specification 3.6.7.2 with an expanded specification that adds requirements which ensure operability of the hydrogen ignition system. Changes to the Limiting Condition for Operation require at least two igniter assemblies in each enclosed area in the containment to be operable as well as all igniter assemblies adjacent to any inoperable igniter assembly in each open area in the containment and drywell. Proposed changes to the Action Statements are provided to coincide with changes to the Limiting Condition for Operations. Proposed changes to Surveillance Requirement 4.6.7.2 are made to demonstrate operability of the hydrogen igniters required operable by the Limiting Condition for Operation. New Surveillance Requirements 4.6.7.2.a.1 and 4.6.7.2.a.2 require energizing the supply breakers at least once every 92 days and verifying a visible glow from each normally accessible igniter assembly in the containment and verifying that each circuit of each containment and drywell hydrogen igniter subsystem is conducting sufficient current to energize the minimum number of igniter assemblies required as specified on new Table 4.6.7.2-1. New Surveillance Requirement 4.6.7.2.b requires, at every Cold Shutdown but no more frequently than once every 92 days, that each normally inaccessible igniter assembly is verified operable by energizing the supply breakers and verifying a visible glow from the glow plugs. New Table 3.6.7.2-1 lists the hydrogen igniters by electrical division and by circuits within each division. New Table 3.6.7.2-2 lists the hydrogen igniters by electrical division/circuit, elevation, azimuth, and distance from the centerline of the reactor. New Table 3.6.7.2-2 also lists those igniters in normally accessible, inaccessible, open or enclosed areas within the containment and/or drywell.

The proposed change to the hydrogen igniter specification follows the staff's guidance to ensure operability of the system. The proposed changes to the Limiting Condition for Operation, Action Statements, and Surveillance Requirements address igniter assemblies in both enclosed and open areas to ensure that all required areas have operable igniters. These changes assure hydrogen igniter system operability and address system design by requiring the normally inaccessible assemblies in the drywell and containment to have a visible glow verification during Cold Shutdown. The licensee indicates that these igniters are inaccessible due to high levels of radiation and/or high temperatures in these areas during plant operation; however, operability is verified for the minimum number required per circuit by electrical current checks at least once per 92 days. The new tables improve the Technical Specification by providing tabulations of igniter location, electrical division and circuit, and minimum number required for each circuit. The proposed changes provide an increase in safety through more stringent requirements than those currently in the Technical Specification and are consistent with the licensing basis. Therefore, the staff finds the proposed changes to be acceptable.

3.7.1.1 Standby Service Water System (TSPS 129, 173)

Limiting Condition for Operating, page 3/4 7-1
Surveillance Requirement, page 3/4 7-2

The proposed revision concerns changes to the standby service water (SSW) system. The change to Technical Specification 3.7.1.1.b is to delete the reference to specific equipment and replace it with a more general reference to "associated plant equipment" and to insert phrases identifying the SSW subsystems required for each Operational Condition. Also, "Operational Condition has been capitalized in Action Statement 3.7.1.1.e to be consistent with the standard format Technical Specifications. As presently written, Specification 3.7.1.1.b requires two independent SSW subsystems to be Operable under all Operational Conditions. The revision provides clarification by requiring two SSW subsystems to be Operable in Operational Conditions 1, 2, and 3. For Operational Conditions 4, 5, and *, the revision requires the SSW subsystem to be Operable consistent with the requirements of Technical Specifications 3.4.9.2, 3.5.2, 3.8.1.2, 3.9.11.1, or 3.9.11.2. The change to Surveillance Requirement 4.7.1.1 involves terminology corrections to reflect that only the SSW subsystem(s) required Operable by Technical Specification 3.7.1.1 are required to be demonstrated Operable by the Surveillance Requirements. The changes provide consistency with the as-built plant and, as such, are consistent with the plant safety analysis and will not adversely impact plant safety. Therefore, the staff concludes they are acceptable.

This proposed revision also concerns changing the SSW system Technical Specification 3.7.1.1 to add Action Statement 3.7.1.1.f to Service Water System Technical Specification 3/4.7.1. This change is to ensure that in all Operational Conditions, when an SSW subsystem is declared inoperable, the associated diesel generator will also be declared inoperable and the Actions Statement required by Technical Specification 3.8.1.1 and 3.8.1.2 will be taken. This proposed change increases safety in that it represents an additional restriction not presently contained in the Technical Specifications. Therefore, the staff concludes this change is acceptable.

3.7.1.2 High Pressure Core Spray Service Water System (TSPS 094, 287)

(1) Limiting Condition for Operation, page 3/4 7-3

This proposed change concerns a change to the high-pressure core spray (HPCS) service water system Action Statement to require that the HPCS diesel generator as well as the HPCS system be declared inoperable when the HPCS service water system is declared inoperable, so that the Action Statements in Technical Specification 3.8.1.1 or 3.8.1.2 will be taken. This change is an improvement to safety in that it represents an additional restriction not presently contained in the Technical Specifications. Therefore, the staff concludes it is acceptable.

An additional change to this specification concerns changes to the HPCS service water system Technical Specification 4.7.1.2 surveillance requirements. The change revises the surveillance requirement to read "The HPCS service water system shall be demonstrated Operable:" with the remainder of the present requirement becoming Surveillance Requirement 4.7.1.2.a. This change also adds Surveillance Requirement 4.7.1.2.b, "At least once per 18 months during shutdown by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on a service water actuation test signal." "Each automatic valve servicing safety-related equipment" refers to the HPCS service water pump discharge valve. These changes result in a more stringent Surveillance Requirement than is presently in the Technical Specifications, and are, therefore, acceptable.

3.7.4 Snubbers (TSPS 006)

> Limiting Condition for Operation, pages 3/4 7-9 and 3/4 7-14 through 3/4 7-25
> Surveillance Requirements, pages 3/4 7-9 through 3/4 7-13
> Bases, pages 3/4 7-2
> Administrative Controls, page 6-23

This proposed change is a general revision to Technical Specification 3/4.7.4 concerning snubbers. This proposed revision follows the intent of the guidelines provided in NRC Generic Letter 84-13 dated May 3, 1984, that allows deletion of the snubber tables provided the associated Technical Specification is modified to specify which snubbers are required to be operable. The proposed change to the limiting condition for operation and applicability requirements for Technical Specification 3/4.7.4 specifies which snubbers are required operable for all plant conditions. Changes proposed to Surveillance Requirement 4.7.4 include additional restrictions not currently included in the Technical Specifications, clarifications, and changes to provide consistency with staff requirements. The changes to Bases 3/4.7.4 are proposed to provide consistency with the subject Technical Specification. Technical Specification 6.10.2.1 is revised to delete reference to the snubber tables that are deleted from Technical Specification 3/4.7.4 by this change proposal.

Some changes have been made in the licensee's proposed specification. The proposed specification limits the accelerated inspection schedule, based on the number of inoperable snubbers found in the visual inspection, to snubbers of the same design type in the system on which the inoperable snubber was found. One modification consists of making the accelerated schedule applicable to snubbers of the same design type in all systems. Another modification moves the statement regarding applicability to non-safety related systems from the Bases to the Limiting Condition for Operation. Another modification makes the retesting due to failure of test equipment applicable to all three functional tests instead of just one.

The staff concludes that the proposed changes, as modified, follow the NRC guidelines and are, therefore, acceptable.

3.7.6 Fire Suppression Systems and
3.7.7 Fire Rated Assemblies (TSPS 070, 071, 072, 131, 203, 223, 244, 245, 277, 338)

Limiting Condition for Operation, pages 3/4 7-28, 3/4 7-31, 3/4 7-33 3/4 7-35, 3/4 7-36, 3/4 7-37, 3/4 7-38, 3/4 7-39, 3/4 7-41

Surveillance Requirements, pages 3/4 7-28 3/4 7-29, 3/4 7-32, 3/4 7-35, 3/4 7-41

Bases, page B 3/4 7-3

The proposed changes to the Technical Specifications for fire suppression systems are as follows:

(1) Spray and/or Sprinkler Systems, Technical Specification 3.7.6.2.

This proposed change corrects an error in a sprinkler identification number and clarifies that this sprinkler system is shared between Units 1 and 2 to reflect as-built conditions and design intent.

(2) Halon Systems, Technical Specification 3.7.6.4

The proposed change to Technical Specification Surveillance Requirement 4.7.6.4.a will delete the hazard area selector valves F497G and F497H from the requirement to verify valve position. All valves in the halon flow paths are totally enclosed, nitrogen pressure or explosive pin actuated valves that cannot be manually manipulated or visually verified to be in the correct position. In addition, operation of any of the valves would actuate the system, which is contrary to the intent of the subject surveillance requirement. All other valves in the halon flow path except valves F497G and F497H can be indirectly verified to be in their correct position by measurement of halon tank pressure since a failed or leaking valve would bleed off the affected tank. The proposed change will make the surveillance consistent with the as-built plant.

(3) Spray and/or Sprinkler Systems, Technical Specification 3.7.6.2

A revision to the subject Technical Specification is proposed that will add a footnote and appropriate reference to indicate that the areas listed for the auxiliary building, control building, and fire pump house are protected by wet pipe sprinkler systems. The diesel generator building is shown to be protected by pre-action sprinkler systems. The proposed changes reflect the as-built plant design.

(4) CO_2 Storage Tank Level, Technical Specification 3.7.6.3

The proposed change increases the minimum CO_2 storage tank level requirement in Surveillance Requirement 4.7.6.3.2.a. from 50% to 60%. The design requirement for the CO_2 storage tank is to provide sufficient capacity for double-shot coverage for the largest room covered in addition to one main generator purge. The proposed change makes the CO_2 storage tank level consistent with the as-built plant. This proposed change is an improvement to plant safety since it applies stricter requirements than the existing Technical Specification.

(5) Fire Rated Assemblies, Technical Specification 3.7.7

This proposed change to Surveillance Requirement 4.7.7.1.c requires that sample selection of penetration seals be such that each penetration seal is inspected at least once during each 15-year period. The present specification requires at least 10% of each type of sealed penetration to be inspected at least once every 18 months but does not require each penetration to be inspected at least once every 15 years. This change is an improvement to plant safety in that it constitutes a more stringent surveillance requirement than is presently in the Technical Specifications.

(6) Deletion of Special Reporting Requirements for Fire Protection, Technical Specifications 3.7.6 and 3.7.7

The proposed revision to delete all references to special reporting requirements for inoperable components of fire suppression systems is in response to NRC Division of Engineering recommendations as found in memorandum from V. Benaroya to C. O. Thomas, "Grand Gulf Nuclear Station Unit 1 - Technical Specifications," dated November 7, 1983. The deletion of the special reporting requirements does not constitute a relaxation of conditions required for safe operation in that the subject Technical Specifications retain all necessary corrective actions to ensure that the plant is operated safely. The revision therefore does not adversely impact plant safety and makes the subject Technical Specifications more easily understood.

(7) Spray and/or Sprinkler Systems, Technical Specification 3.7.6.2

This proposed change revises Surveillance Requirement 4.7.6.2.c and Bases 3/4.7.6 to include visual inspection to ensure that spray areas and patterns are not obstructed by temporary structures or objects. This change is an improvement to plant safety in that it constitutes an additional control not presently included in the Technical Specifications.

(8) Fire Suppression System Surveillance Requirements, Technical Specification 3.7.6.1

This proposed change provides an additional surviellance requirement for the fire suppression water system. Surveillance Requirement 4.7.6.1.1.e was added to ensure that a system flush is performed at least once every 12 months. This proposed change improves plant safety because it provides a more stringent surveillance requirement.

(9) Fire Hose Stations, Technical Specification Table 3.7.6.5-1

This proposed change adds four fire hose stations to the list of fire hose stations that may be relied on to confine and extinguish fires occurring in a portion of the facility where safety-related equipment is located. The proposed change is appropriate because safety-related cables pass through the areas covered by these hose stations. These fire host stations are installed in the plant and this change is proposed to reflect as-built plant design and usage. This change is an improvement to plant safety in that it increases that amount of fire protection equipment included in the Technical Specification.

3.7.8 Area Temperature Monitoring (TSPS 132)

Limiting Condition for Operation, page 3/4 7-44

This proposed change concerns changing the control room temperature limits listed in Table 3.7.8-1 from 77°F to 90°F and deletes the "Equipment Not Operating" column and the "Equipment Operating" heading.

The 77°F control room temperature limit is increased to 90°F because the present limit of 77°F is derived from human factors considerations rather than equipment qualification data. NUREG-0700, "Guidelines for Control Room Design Reviews," requires that the control room HVAC system be capable of maintaining the dry bulb temperaure between 73°F and 77°F. This is a system performance standard for maintaining the comfort zone for personnel occupancy. The control room temperature limit of 90°F is based on a review of the control room equipment qualification data sheets. The lowest environmental qualification temperature of any equipment in the control room was found to be 90°F. Control room temperatures exceeding this limit for more than 8 hours requires an evaluation of the impact on the qualified life of the affected equipment as required by the present technical specification Action Statement. The use of the lowest qualification temperature for the control room is consistent with the limits established for other areas listed in this table. Because the existing control room Technical Specification limit of 77°F is based on human factors performance standard and not an equipment performance standard, as is the intent of Table 3.7.8-1, a change to 90°F is acceptable, because the NRC staff concern is with equipment performance.

The change to delete the present area temperature limits when equipment is not operating is made because they may not be the limiting temperature for the affected areas. The change to a single temperature limit will also eliminate confusing and possibly conflicting requirements.

The change to delete the table heading "Equipment Operating" is administrative because these temperature limits will apply at all times.

These changes are for clarification and to bring the control room temperature limits in line with the method for determining the temperature limits elsewhere in the plant. Based on its review, the staff concludes that the proposed change is acceptable.

3.7.9 Spent Fuel Storage Pool Temperature (TSPS 058, 826)

Limiting Condition for Operation, page 3/4 7-45
Surveillance Requirement, page 3/4 7-45

This proposed change concerns a change to the spent fuel pool temperature Action Statement and to the Surveillance Requirements. The Action Statement for exceeding the spent fuel storage pool (SFSP) temperature limit requires that the pool temperature be reduced to less than or equal to 150°F within 8 hours, or be in at least Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours. The change to this Action Statement is to replace the requirement for Shutdown with a requirement to submit a Special Report to the Commission whenever the 150°F limit has been exceeded for longer than 72 hours. This is acceptable because the temperature of the SFSP has no effect on the safety of operation of the plant, and plant Shutdown would not affect the temperature of the SFSP or aid in cooling it any faster. Surveillance Requirement 4.7.9.2 is also added to ensure fuel pool cooling pump Operability. These changes constitute an improvement to the Technical Specification without adversely affecting plant safety.

In addition, this proposed revision concerns changes to the spent fuel pool temperature Surveillance Requirements to Technical Specification 4.7.9.1 to allow other acceptable methods of determining the bulk pool temperature. This change is consistent with the safety analysis and is a clarification of the intent of the surveillance requirement.

Based on its review, the staff concludes the proposed changes are acceptable.

3.7.10 Embankment Stability (TSPS 133)

Limiting Condition for Operation, page 3/4 7-46
Bases, page B 3/4 7-4

This change would relax the Limiting Condition for Operation for the specification that ensures Culvert No. 1 on the plant site will not be blocked. The required stability of the downstream slope of the access road embankment and the limit on the maximum permissible blockage of Culvert No. 1 are intended to ensire that Culvert No. 1 is always functional, because, in the event the culvert is blocked, flooding of the plant and safety-related facilities could occur during a probable maximum flood (PMF) event. The current specification requires action to verify slope stability and clean the culvert, with blockage of 15% of its own sectional area. The change would not require this until blockage was 45%. The current specification requires the plant to be shut down within 6 hours in accordance with Technical Specifications 3.0.3 and 3.0.4 if the Limiting Condition for Operation is not met. The change would allow 7 days for cleaning the culvert.

The licensee requested deletion of Technical Specification 3/4.7.10 and addition of a requirement for an embankment stability verification program in Technical Specification 6.0, Administrative Controls. The staff concludes that the change from a specification to a program is not acceptable. However, the staff in its evaluation of the issue has concluded that the percent blockage for Culvert No. 1 can be changed from 15% to 45%.

The existing Technical Specification 3/4.7.10 was proposed by the licensee in AECM-83/0370 dated June 29, 1983, in order to implement a requirement in Supplement 1 to the Grand Gulf Safety Evaluation Report (SER). The staff concludes that the change from a specification to a program is not acceptable.

When the SER was issued, the design basis flood level for the main plant area was the probable maximum flood on drainage basin B at elevation 132.8 ft MSL (mean sea level). The flood evaluation for stream B assumed that Culvert No. 1 was 45% blocked by debris. The possibility of blockage of Culvert No. 1 is the basis for the existing Technical Specification and Limiting Condition for Operation. The requirement that the culvert must be cleaned out when more than 15% blocked provided assurance to the staff that the culvert blockage would not exceed the assumed 45% value during a large flood.

Subsequent to the SER issuance, the staff advised that changes made to the site topography in the power block area in the final stages of construction had adversely affected the runoff of water during the site's postulated probable maximum precipitation (PMP) event. This concern was identified in a final review of site drainage acceptability and was reported to NRC Region II in accordance with 10 CFR 50.55(e) on May 10, 1982, as potentially reportable deficiency. These changes in site topography resulted in potential floodwaters during the PMP event as high as elevation 133 ft 5 in. Previously, the local PMP event did not exceed elevation 133 ft MSL, the entrance floor elevation to some safety-related buildings.

The licensee proposed, as a resolution, a permanent modification in the form of mechanical seals on affected doors that would extend 6 in. above the PMP standing water height or about elevation 134 ft MSL. The staff reviewed and accepted this proposal, and the seals have been installed as reported in Inspection Report 50-416/84-18 dated June 5, 1984.

During a meeting between the staff and licensee's representatives on March 28, 1984, the licensee was advised that in order to delete Technical Specification 3/4.7.10, they must submit an analysis to show that a PMF on stream B, assuming 100% blockage of Culvert No. 1, would not jeopardize the shutdown capability of the plant.

The results of a preliminary analysis by the staff have shown that the PMF on stream B, with 100% blockage of Culvert No. 1, would produce a flood level of about 134.3 ft MSL for a short period of time. The analysis also showed that once the temporary buildings located west of the main plant area are removed, the flood level would not exceed elevation 134 ft MSL.

Therefore, the staff concludes that the Technical Specification should remain in effect, as modified, until the licensee can (1) show that the PMF flood level on stream B, with 100% blockage of Culvert No. 1, does not exceed elevation 134 ft MSL; or (2) the seepage through the doors when the flood level is above elevation 134.0 ft MSL can be effectively controlled or contained and will not impair the ability to safely shut down the plant.

With the additional foot of protection provided by the door seals, the PMF on stream B can now be conveyed below elevation 134 ft MSL with as much as 80 or 90% blockage of Culvert No. 1. Thus the staff can relax the 15% blockage requirement to 45% and allow 7 days for cleanout. The title of the specification has been changed to "Flood Protection" to denote its primary purpose.

<u>3.8.1.1 AC Source - Operating</u> (TSPS 007, 026, 043, 145, 174, 175, 342)

Limiting Condition for Operation, pages 3/4 9-1, 3/4 8-2
Surveillance Requirements, pages 3/4 9-4 through 3/4 8-7
Bases, page 3/4 8-1

The proposed changes follow:

(1) In Surveillance Requirement 4.8.1.1.2.c regarding the collection of data on fuel oil sampling, the standard year referenced for ASTM D270 is in error. The requirement should read "in accordance with ASTM D270-1965 (reapproved 1980)."

(2) The Surveillance Requirement for verification of the diesel generator air start capacity would be deleted. The Surveillance Requirement requires periodic verification of the adequate capacity (fire start capability) of the diesel generator over the lifetime of the plant. This requirement needs to be done only once during the preoperational tests. This has been done by the licensee. Therefore, the request to delete the surveillance requirement is acceptable.

(3) A footnote would be added to Action Statements a, b, d, and e of Technical Specification 3.8.1.1 to clarify when the action must be taken. The proposed change is to add footnote * to Action Statements a, b, d, and e to clearly indicate that diesel generator 13 testing per Surveillance Requirement 4.8.1.1.2.a.4 is required only when the HPCS system is operable as diesel generator 13 supplies power only to the HPCS system. This change clarifies the specification and prevents unnecessary testing of diesel generator 13 and, therefore, is acceptable.

(4) Surveillance Requirement 4.8.1.1.2.d.6 regarding tests with a simulated loss of the diesel generator, with offsite power not available, would be deleted. Generic Letter 83-30 provided a revision to the Surveillance Requirements for diesel generator testing which deleted section 4.8.1.1.2.d.6. The letter further stated that licensees may propose amendments to the Technical Specification to delete the subject section. Therefore, this change is in accordance with Generic Letter 83-30 and is acceptable.

(5) Surveillance Requirement 4.8.1.1.2.d.14 regarding verification of functioning of the transfer of fuel oil from the storage tank to the day tank would be deleted. This surveillance requirement applies only to plants with cross-connections between the diesel generator fuel oil systems. Because Grand Gulf does not have this type of design, this requirement is not applicable and should be deleted.

(6) This change would increase the time allowed for demonstrating diesel generator operability when an offsite circuit or diesel generator is inoperable. The Action Statements for Technical Specification 3.8.1.1 requires surveillance tests to be performed to demonstrate redundant equipment to be operable when diesel generator(s) and/or offsite power circuit(s) are determined to be inoperable. Difficulties have been encountered in performing the surveillances required by the Action Statements within the specified time limits. The Grand Gulf Technical Specifications presently allow 1 hour per diesel generator.
The Technical Specification changes would allow 2 hours in Action Statements a, d, and e of Technical Specification 3 8.1.1. These changes are in accordance with the criterion and, therefore, are acceptable.

(7) A clarification would be made to the Bases regarding fuel oil quantity specification. This change was requested to clarify that the fuel oil quantity specified in the Technical Specification means usable fuel rather than minimum fuel in the storage tanks. The minimum fuel oil level in the storage tanks would include the non-usable portion at the bottom of the tank, whereas the usable fuel is that fuel oil available to the engine for 7 days of operation. The staff finds the clarification acceptable.

3.8.1.2 AC Power Sources - Shutdown (TSPS 060, 176)

Limiting Condition for Operation, page 3/4 8-9
Surveillance Requirement, page 3/4 8-9

This proposed change concerns suspending crane operations over the upper containment pool in addition to over the spent fuel pool whenever all offsite circuits are inoperable and/or with diesel generators 11 and 12 inoperative. The proposed change improves safety in that it adds a requirement to the specifications. The staff concludes that the change is acceptable.

Additional changes proposed for this specification provide a clarification to the operability requirements. The following changes are proposed:

(1) Change the "and/or" in Technical Specification 3.8.1.2.b to "or."

(2) Change the "and/or" in Action Statement a. line 2 to "and."

(3) Delete the phrase "of the above required AC electrical power sources" from Action Statement a.

The proposed changes clarify the specification; the revised limiting condition and action requirement is a more straightforward statement of the intended requirements. Based on its evaluation, the staff finds the proposed changes acceptable.

3.8.2.1 DC Sources - Operating (TSPS 227)

Surveillance Requirement, page 3/4 8-11

This proposed change revised Surveillance Requirement for Divisions 1 and 2 to require that the battery charger supply 400 amperes at a minimum of 125 volts for at least 10 hours for Division 1 and 2. For Division 3, the Surveillance Requirement is changed to require that the battery charger is to supply 50 amperes at minimum of 125 volts for 4 hours. These changes provide surveillance testing requirements for the designed capacity of battery chargers and are in conformance with associated requirements in FSAR. Therefore, these changes are acceptable.

3.8.4.1 Electrical Equipment Protective Devices (TSPS 302)

Limiting Condition for Operation, pages 3/4 8-21 and 3/4 8-37a through 3/4 8-37c
Surveillance Requirement, page 3/4 8-20

The change to Surveillance Requirement 4.8.4.1 Section a.2 is proposed to clarify that when circuit breakers are inoperable, they shall be restored to Operable status before resuming operation of the affected equipment. As presently worded, the Surveillance Requirement could be misinterpreted to imply that plant operation could not resume if a circuit breaker were found inoperable. The staff concurs with the proposed addition of the phrase "of the affected equipment" after "resuming operation" in Section 4.8.4.1.a.2.

The 125 VDC and 120 VAC circuit breakers have been added to Table 3.8.4.1-1 only to ensure that appropriate surveillances are performed to detect degradation of these devices over the life of the plant. These surveillances are required because of the possibility of these circuits developing sufficient fault current to cause damage to containment penetrations. The response time of the 480 VAC circuit breakers in Technical Specification Table 3.8.4.1-1 is increased from 0.05 second to 0.07 second. This change is necessary to ensure that the response time is long enough to include all breaker trips when subjected to a test current of 120% of the instantaneous trip setpoint. A response time of 0.07 second is much lower than the maximum fault current versus time limit of 0.52 second as shown in FSAR Figure 40.5-7 (No. 4/0 AWG penetration cable time - current characteristic curves). The addition of the 125 VDC and 120 VAC circuit breakers provides an improvement to plant safety in that it clarifies the intent of the surveillance and adds more stringent requirements to the Technical Specification. The change in response time ensures a satisfactory breaker response test that must not be exceeded to avoid possible degradation of cable. Therefore, these changes are acceptable.

3.8.4.2 Motor Operated Valves (MOV) Thermal Overload Protection (TSPS 061, 136, 137, 179, 228)

Limiting Condition for Operation, pages 3/4 8-38, 3/4 8-39, 3/4 8-43
Surveillance Requirement, page 3/4 8-38

(1) Technical Specification 4.8.4.2.1.a requires that a channel functional test be conducted once every 92 days on the bypass circuitry for the MOV thermal overloads that are normally in force during plant operation and bypassed during an accident condition. This would require the injection of a simulated signal into the LOCA signal transmitter, causing a false ECCS actual signal. This would result in the disruption of plant operation once every 92 days. The intent of the subject surveillance is to test only the individual valve bypass circuitry.

The proposed change is a revision to Surveillance Requirement 4.8.4.2.1.a. that allows a full channel test once every 18 months and a test of individual valve bypass circuitry once every 92 days. The Technical Specification change is proposed as follows:

"1. At least once per 92 days for the individual valve bypass circuitry.

2. At least once per 18 months for the active channel.",

To prevent spurious interruption of plant operation once every 92 days and is, therefore, acceptable.

(2) The licensee also proposed that the phrase "take administrative action to" be deleted from Technical Specification 3.8.4.2, which specifies action to be taken when thermal overload protection of a safety-related valve is found to be not operable or not bypassed either continuously or only under accident conditions. The phrase is considered to be redundant and does not improve the intent of the Technical Specification. Because this is a purely administrative change to clarify the Technical Specification and does not alter the actions to be taken, the staff concludes that the

proposed action is acceptable on the basis of technical and safety considerations.

(3) Another proposed change corrects the listing for valves Q1M71F593A and Q1P41F018 to be Q1M71F593 and Q1P41F018A, respectively, in Table 3.8.4.2-1, and is, therefore, acceptable.

(4) Technical Specification Table 3.8.4.2-1 lists "Valve on Turbine Q1E51C002" in the valve number column. This designation does not clearly define a specific RCIC system valve. The proposed change revises valve number "Valve on Turbine Q1E51C002" to "RCIC Trip and Throttle Valve on Turbine Q1E51C002" for clarification and is, therefore, acceptable.

(5) The "A" and "B" designations on the valve numbers for the drywell monitoring system, RCIC system and reactor coolant system in Table 3.8.4.2-1 are erroneously indicated as part of the valves' number. Proposed changes are made to delete the "A" and "B" designation from the valve number for correction. These changes are, therefore, acceptable.

3.8.4.3 Reactor Protection System Electric Power Monitoring (TSPS 180)

Surveillance Requirement, page 3/4 8-46
Bases, page B 3/4 8-3

This proposed change revises the overvoltage and undervoltage setpoint values for Bus A and Bus B of the protective instrumentation in the Surveillance Requirement. The changes are made to reflect system design as determined by voltage drop analysis for precise overvoltage and undervoltage setpoints. These changes: (1) "Over-voltage bus A $\leq$ 132.9 VAC, Bus B $\leq$ 133.0" from "Over-voltage $\leq$ 132 VAC," (2) "Under-voltage Bus A $\geq$ 115.0 VAC, Bus B $\geq$ 115.9 VAC," from "Under-voltage $\geq$ 117 VAC," and (3) "Under-frequency Bus A $\geq$ 57 Hz, Bus B $\geq$ 57 Hz" from "Under-frequency $\geq$ 57 Hz" are improvements to plant operation and are made to prevent unnecessary or spurious trips to the reactor protection system power supply. Therefore, these changes are acceptable.

3.9.2 Refueling Operations (TSPS 251, 323)

3.10.3 Shutdown Margin Demonstrations

Limiting Condition for Operation, page 3/4 9-3
Surveillance Requirement, pages 3/4 9-3, 3/4 9-4, 3/4 10-3

(1) The proposed changes revise Technical Specification 3.9.2 to require that the shorting links be removed from the reactor trip system circuitry before and during any time a control rod is withdrawn, unless adequate shutdown margin has been demonstrated. In addition, the proposed change revises Technical Specification 3/4.10.3 to require that the shorting links be removed during shutdown margin demonstrations, and either the rod pattern control system be operable or a second operator verifies the procedure.

The licensee has stated that the proposed changes are to incorporate NSSS vendor (GE) recommendations to prevent inadvertent criticality when the reactor head is removed during shutdown margin demonstrations. Presently, the Technical Specifications permit shutdown demonstrations to be performed with either the reactor trip system shorting links removed or with the rod pattern control system operable. Removal of the shorting links enables a reactor scram (control rod insertion) on sensing a (noncoincident) source-range monitor (SRM) high-high signal. The rod pattern control system includes interlocks that prevent the withdrawal of more than one rod at a time. During the performance of shutdown margin demonstrations, procedures require the bypass of the rod pattern control system. The proposed change requires that during this bypass the shorting links be removed and a second licensed operator or other qualified person verify procedural conformance. Based on its review, the NRC staff finds that the proposed changes provide a sufficiently conservative set of requirements during shutdown margin demonstrations. Therefore, the staff finds the proposed change acceptable.

(2) The proposed change in Surveillance Requirement 4.9.2 would reduce the minimum source range monitor count rate required for operability from 3 counts per second (cps) to 0.7 cps. This change has been approved for other BWRs (e.g., LaSalle, Shoreham) and is acceptable for Grand Gulf provided that the following footnote is added to the bottom of page 3/4.9-4:

"*Provided signal to noise ratio $\geq$ 2, otherwise 3 cps"

3.9.3 Control Rod Position (TSPS 280)

Limiting Condition for Operation, page 3/4 9-5

The proposed change revises Technical Specification 3.9.3 by moving a permissible exception to the Limiting Condition for Operation (LCO) from the Action Statement to a footnote (*). The LCO requires that all control rods be inserted in Operational Condition 5, during Core Alterations. The Action Statement requires that during conditions when all control rods are not inserted, Core Alterations must be suspended. The permissible exception allows one control rod to be withdrawn under control of the reactor mode switch (Refuel Position) one-rod-out interlock. The licensee has stated that the proposed change provides relief from an administrative burden encumbered by entering an Action during single control rod withdrawal for function testing, subcriticality checks, and instrumentation response checks. Based on its review, the NRC staff finds that the proposed change does not alter any existing requirements. Therefore, the staff finds the proposed change acceptable.

3.9.6 Refueling Equipment (TSPS 035)

Limiting Condition for Operation, page 3/4 9-8
Surveillance Requirement, page 3/4 9-8
Bases, page B 3/4 9-1

The staff reviewed the proposed change to Technical Specification 3.9.6 that would update the specifications to be consistent with the BWR-6 refueling equipment provided at Grand Gulf. The original specification, entitled, "Refueling Platform," follows the BWR Standard Technical Specifications for plants built before development of the BWR-6 design. The proposed specification is entitled, "Refueling Equipment" and has three separate parts to cover the refueling platform, auxiliary platform, and fuel handling platform. This is consistent with the BWR-6 design at Grand Gulf.

The proposed changes are primarily necessary to account for equipment changes, procedural changes, and differences in nomenclature for the BWR-6 design because the original specifications pre-date BWR-6 plants. The proposed changes to the Technical Specification retains the applicable technical requirements of the existing specification for the refueling platform, and the applicability has been expanded to include the auxiliary platform and fuel handling platform. The Limiting Conditions for Operation and associated action requirements identified in the original specification for the refueling platform still apply and are also applicable for the fuel handling platform and auxiliary platform.

The only significant proposed change is the deletion of a Surveillance Requirement for demonstrating the up-travel mechanical stop function on the refueling platform auxiliary hoist within 7 days before the start of handling operations within the reactor vessel. The function of the up-travel mechanical stop function on the auxiliary hoist was to ensure that irradiated fuel would be adequately shielded during handling operations. The proposed specifications will prohibit the handling of irradiated fuel assemblies by the auxiliary hoist. Therefore, the up-travel mechanical stop surveillance requirement is not needed.

Based on its review of the proposed changes, the staff concludes that they adequately reflect the Grand Gulf refueling equipment design and meet the technical requirements and intent of the BWR Standard Technical Specifications. The staff further concludes that the proposed changes do not result in a significant reduction in safety margin because they were changes necessary to reflect the actual Grand Gulf Unit 1 design, and are, therefore, acceptable.

The Bases for this specification has also been changed to make it consistent with the refueling equipment used at Grand Gulf. The change reflects that only the main hoist of either the refueling platform or the fuel handling platform will be used to handle irradiated fuel assemblies. Additionally, the Bases are revised to indicate that all platform hoists have sufficient load capacity for handling fuel assemblies and/or control rods. The staff concludes that the changes were necessary to make them consistent with the refueling equipment and are, therefore, acceptable.

3.9.10.2 Multiple Control Rod Removal (TSPS 307)

Limiting Condition for Operation, page 3/4 9-14
Surveillance Requirement, page 3/4 9-15

The change to Technical Specification 3.9.10.2 would add the requirement that all fuel loading operations be suspended unless all control rods are inserted into the core. Because this represents an added conservatism in the Technical Specifications, the staff finds it acceptable.

3.9.12 Horizontal Fuel Transfer System (TSPS 267)

Limiting Condition for Operation, page 3/4 9-18
Surveillance Requirement, page 3/4 9-18

This proposed change concerns a change to the horizontal fuel transfer system to add the room number and elevation to Limiting Condition for Operation

3.9.12.a and to add a corresponding Surveillance Requirement to verify periodically that the room through which the transfer system penetrates is sealed during transfer system operation. Existing Surveillance Requirements "a" and "b" are redesignated "b" and "c", respectively. Access to this area during fuel transfer operations could result in personnel exposure in excess of 10 CFR 20 requirements. The revision will ensure compliance with Technical Specification 3.9.12.a and represent an additional control currently not present in the Technical Specifications. Therefore, the staff concludes that the change is acceptable.

3.10.2 Rod Pattern Control System (TSPS 183)

Surveillance Requirement, page 3/4 10-2

This change to Technical Specification 4.10.2 is made to clarify the Surveillance Requirements to be met when the rod pattern control system (RPCS) is bypassed and to specify the frequency requirements for such surveillance. Because this represents an improvement to this Technical Specification, the staff finds it acceptable.

3.11 Radioactive Effluents

3.12 Radiological Environmental Monitoring (TSPS 036, 085, 086, 088, 089, 090, 092, 135, 138, 190, 191, 192, 194, 249)

Pages 1-6, 1-8, 3/4 11-1, 3/4 11-3 through 3/4 11-20, 3/4 12-1, 3/4 12-3 through 3/4 12-7, 3/4 12-10 through 3/4 12-12

One of the proposed changes to the radioactive gaseous waste sampling and analysis program (Table 4.11.2.1.2-1) restructures the table to more clearly reflect the requirements for sampling radioactive gaseous wastes and to be consistent with the as-built plant design. The restructured table includes two functional release-type categories (continuous and intermittent releases). The licensee added two additional grab sample points for gaseous release, the fuel handling area ventilation exhaust point, and the radwaste building ventilation exhaust point. This change provides an additional monitoring requirement not included in the current technical specifications. The tritium sampling and analysis requirements following startup from cold shutdown or after a 15% or greater rated thermal power change are deleted. Instead, the tritium analysis is required for monthly grab samples from all building ventilation exhausts, including radwaste building and fuel handling area ventilation exhausts. The reference to Note (f) in Section D of the table is a typographical error and therefore has been eliminated (there was no Note (f)).

Another proposed change to the specification for radioactive effluent from the main condenser deletes specific isotopes from the Limiting Condition for Operation, the associated Action Statement, and Surveillance Requirement 4.11.2.7.2 to make the Technical Specification applicable to all noble gases that are existent after a 30-minute decay period. This allows for the decay of the short-lived noble gas isotopes, which are not considered significant in effluent release considerations. The main condenser air ejector effluent monitoring system uses gamma scintillation detectors to measure the Kr-85m, 87, and 88, and Xe-133, 135, and 138 contribution after 30 minutes of decay.

A footnote is added to the applicability of Technical Specification 3/4.11.2.7, for Operational Conditions 2 and 3, to reflect applicability only during operation of the main condenser air ejector. An additional footnote is provided for Surveillance Requirement 4.11.2.7.2 to indicate that the provisions of Technical Specification 4.0.4 do not apply. These changes are for clarifications indicating that there is no need to monitor air ejector effluent if the air ejectors are not in service.

All of the remaining proposed Technical Specifications changes fall into this category. They consist of a general upgrading of the current Grand Gulf Unit 1 Radiological Effluent Technical Specifications (RETS) to be consistent with Draft 7, Revision 3 of NUREG-0473, the current guidance being used in the program for implementing RETS in all operating reactors. In each specification, the changes requested make the specification consistent with NRC guidance. Thus, the licensee's proposed changes in their RETS meet the intent of NUREG-0473.

Based on its evaluation, the staff finds that the proposed changes are mostly administrative and clarifying in nature, and are made to reflect the as-built conditions of the radiation monitors and the plant systems. The staff concludes that the proposed changes meet the intent of NUREG-0133 and NUREG-0473, and do not remove or relax any existing requirements related to (1) the effluent radiation monitoring, and (2) the probability or consequences of accidents previously considered in the Grand Gulf SER. The staff finds that the proposed changes are consistent with the Grand Gulf FSAR and SER, and are, therefore, acceptable.

<u>6.0 Administrative Controls</u> (TSPS 052, 063, 093, 095, 096, 101, 106)

(1) By letter dated May 24, 1984, the licensee proposed changes to Technical Specification Figure 6.2.1-1, "Offsite Organization," and Figure 6.2.2-1, "Unit Organization." Except for the organizational position of Chemistry/Radiation Control Superintendent, the staff finds these changes to be acceptable as proposed. The staff's evaluation of the position of Chemistry/Radiation Control Superintendent is provided herein. The staff's evaluation of the remainder of the organizational changes is reported in Supplement 5 of the SER.

    The NRC staff has evaluated these changes for their impact on the Grand Gulf radiation protection program against the criteria in RG 8.8 (Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable), RG 1.8 (Personnel Qualification and Training), and NUREG-0731, "Guidelines for Utility Management and Technical Resources," as provided in Chapter 12 of the Standard Review Plan (NUREG-0800).

    The position of Chemistry/Radiation Control Supvervisor, who is the radiation protection manager (RPM) as described in the FSAR, is impacted by the proposed organizational change in three major areas:

    (a) Radiation protection organizations should be independent from operating pressure in accordance with RG 8.8, Section C.1.b(3), and NUREG-0731, Section II.A.1 (SRP Section 12.5).

(b) Distinct functional areas should be separately supervised and/or managed (i.e., radiation protection should be separate from chemistry in accordance with NUREG-0731, Section II.A.1 (SRP Section 12.5)).

(c) The RPM should be qualified in accordance with RG 1.8 (SRP Section 12.5).

In its review of the licensee's letter of May 24, 1984, and the proposed new organization, the staff noted that this new organization differs markedly from that reviewed and reported in the original SER. The new organization changes the functions (and title) of the manager to whom the RPM reports. Previously, the RPM reported to the Assistant Plant Manager, Nuclear, who had broad responsibilities and functioned as an assistant plant manager, being responsible for operations, maintenance, chemistry, and radiological controls. In the proposed organization, the RPM would report to a new position of Manager, Plant Operations, who would have a considerably reduced scope of responsibilities, primarily operations responsibilities. Under the proposed organization, having the RPM report to the Manager, Plant Operations, whose main function is operational, would not provide the independence from operational pressures that existed in the previous organization's Assistant Plant Manager position. Additionally, no direct access to the General Manager is provided for the RPM. The licensee has offset these problems by appointing the RPM to the Plant Operations Review Committee (PORC) and by establishing procedures enabling direct access of the RPM to the General Manager.

Based on its review, the staff concludes that the Chemistry/Radiation Control Superintendent should be provided with direct access to the General Manager for matters involving radiation protection, and that Figure 6.2.2-1 should be modified to reflect this line of communication. In this manner, the independence of the radiation protection program from operating pressures will be maintained, and direct access to the General Manager will be ensured for the Chemistry/Radiation Control Superintendent.

The continued combination of health physics and chemistry functions in the Chemistry/Radiation Control Superintendent, although not strictly in accordance with SRP positions, remains acceptable as long as the training and qualification commitments and requirements for personnel with combined functions are met, and particularly as long as independence from operating pressures is maintained. This aspect of the proposed change remains acceptable as noted.

Under the revised organization, the RPM is now the Chemistry/Radiation Control Supervisor, which is a level above that of the previous organization. This meets the staff positions in RG 8.8 and NUREG-0731 (SRP Section 12.5) and is, therefore, acceptable.

(2) By letter dated June 22, 1984, the licensee requested changes to the Grand Gulf Technical Specifications, Section 6, Administrative Controls. The staff's evaluation of these proposed changes follows.

One proposed change corrects an original error of omission, regarding the responsibility for reviewing temporary changes to plant procedures within

14 days, and makes position title changes. The staff finds these changes acceptable as reported in Supplement 5 to the SER.

A proposed change to Technical Specifications 6.2.3.4 and 6.3.1 raises the position to which the Independent Safety Engineering Group (ISEG) makes its recommendations. ISEG will report to the Senior Vice President, Nuclear, rather than to the Assistant Vice President, Nuclear Production. The change also permits the use of ISEG of individuals who do not have the engineering and experience qualifications that the required minimum of five engineers would have. As long as the core group of five have met these required qualifications, the staff concurs with the idea that ISEG functions may benefit from additional expertise. Therefore, these are acceptable changes.

A proposed change to Technical Specification 6.5.2.7 would change the review functions of the corporate Safety Review Committee (SRC) so that the SRC could delegate its review functions. This is not acceptable because it would diffuse the attention and direct involvement of the SRC members in the reviews that should be performed by experienced, high-level technical management. Another change related to this specification would also add a review item to the SRC responsibilities to be consistent with the FSAR, and is, therefore, acceptable.

The proposed changes to Technical Specification 6.5 would (1) make an editorial change, (2) make changes in position titles, and (3) add the Technical Engineering Supervisor as a member of the Plant Safety Review Committee. These changes are acceptable.

The licensee has proposed by letter dated September 9, 1983, to include a requirement under Annual Reports page 6-16, to include in the annual reports documentation of all challenges to safety/relief valves. The staff concludes that such reporting will satisfy regulatory requirements and that the change is, therefore, acceptable.

By letter dated June 22, 1984, the licensee proposed a revision to the Technical Specifications to incorporate the recommendations of NRC Generic Letter 83-43 dated December 19, 1983. The Generic Letter provides policy guidance concerning the implementation of Technical Specification changes required as a result of the addition of 10 CFR 50.73, "Licensee Event Report System." The major proposed change is the deletion of Technical Specifications 6.9.1.11, 6.9.1.12, and 6.9.1.13. In addition, the licensee made a complete review of the Grand Gulf Technical Specifications to identify any additional specifications that should be changed. The NRC staff has determined that the proposed changes will make the Technical Specifications consistent with the requirements of 10 CFR 50.73 and therefore concludes that the proposed changes are acceptable.

16.4.2 Miscellaneous Changes

The changes described herein are proposed to correct typographical errors, clarify the text, correct inconsistencies within the specifications and with the FSAR and as-built plant, and incorporate editorial comments. The staff has reviewed proposed changes and determined that they improve the clarity of the specifications and do not significantly change the conditions for operation, action statements, or surveillance requirements. Therefore, the staff concludes that the changes described herein are acceptable.

Index (TSPS 097)

The index will be changed to reflect text changes.

1.0 Definitions (TSPS 085, 091, 093, 164, 167, 225, 249, 818)

Page	Change
1-2 1-6	Change "OPERABLE pursuant to" to "in compliance with the requirements of"
1-3	Correct misspelling
1-4	Add definition of "Member(s) of the Public"
1-6	Add to definition of Process Control Program "burial ground requirements" and "10 CFR Part 61"
1-7	Change "Reportable Occurrence" to "Reportable Event" and change "Specifications 6.9.1.12 and 6.9.1.13" to "Sections 50.73 of 10 CFR Part 50"
	Add "rupture disc" to definition of Secondary Containment Integrity
1-8	Add definitions of Site Boundary and Unrestricted Area; change definition of "Solidification"

2.0 Safety Limits and Limiting Safety System Settings (TSPS 053, 074, 298, 319)

Page	Change
2-2	In the first sentence of the Action statement, delete "after depressurizing the reactor vessel, if required" and add a new sentence "Depressurize the reactor vessel as necessary for ECCS operation"
B 2-2	Change "NEDO-203040" to "NEDO-20340"
B 2-5	Change ASME Boiler and Pressure Vessel Code from "1974 Edition, including Addenda through Summer 1975" to "1971 Edition, including Addenda through Winter 1972"
B 2-8, B 2-9	Add information regarding reactor trip system setpoints for drywell pressure high, turbine control valve fast closure, reactor mode switch shutdown position, and manual scram circuits

3.0 Limiting Conditions for Operation
4.0 Surveillance Requirements

3.1 Reactivity Control Systems (TSPS 051, 154, 156, 157, 255, 313)

Page	Change
3/4 1-7	Change "3.1.3.2.c" to "4.1.3.2.c"
3/4 1-15 B 3/4 1-4	Correct misspellings or grammar
3/4 1-18, 3/4 1-19	Change "the standby liquid control system" to "two standby liquid control system subsystems"
3/4 1-19	Change "Storage tank heaters" to "Storage tank heater"
3/4 1-20	Redraw barely legible Figure 3.1.5-1
B 3/4 1-2	Add information regarding control rod drive system to clarify that "inoperative rods" must be "trippable" and clarify analyses to determine limiting control rod insertion times

3.2 Power Distribution Limits (TSPS 158)

Page	Change
3/4 2-5	Change "always less than or equal to one" to "applied only if less than or equal to one"

3.3 Instrumentation (TSPS 018, 045, 074, 079, 093, 109, 112, 185, 196, 238, 278, 345, 348)

Page	Change
3/4 3-1, 3/4 3-9	Delete from a footnote "With a design providing only one channel per trip system"
3/4 3-4	Add "insertable" before "control rods"
3/4 3-14	Capitalize Operational Condition
3/4 3-19	Change "3.6.5.2-1" to "3.6.6.2-1"
3/4 3-23a	Typographical error
3/4 3-25	Add footnote (b) to the manual initiation circuits for Division 1 and 2 low-pressure coolant injection and low-pressure core spray systems; footnote (b) provides design information
3/4 3-29	Delete an unreferenced footnote

Page	Change
3/4 3-60, 3/4 3-63, 3/4 3-76, 3/4 3-81	Delete the phrase "in lieu of any other report required by Specification 6.9.1"
3/4 3-63	Insert "required" before "meteorological monitoring"
3/4 3-64	Change "Minimum Instruments Operable" to "Minimum Channels Operable"
3/4 3-82, 3/4 3-87	Delete reference to Section 6.9.1.11
3/4 3-82, 3/4 3-87	Capitalize "Offsite Dose Calculation Manual" and change "or" to "and if unsuccessful"
3/4 3-87	Change "these channels" to "applicable channels" and add a footnote to reference specifications for explosive gas and offgas pretreatment monitors
3/4 3-91	Change Action Statement from "otherwise be in at least HOT STANDBY, within 12 hours" to "otherwise be in at least HOT SHUTDOWN within 12 hours, and in COLD SHUTDOWN within the following 24 hours"
3/4 3-93	Change the allowable value of reactor vessel water level-high from "$\leq$ 55.7 inches" to "$\leq$ 54.1 inches"
B 3/4 3-1	Delete the sentence "The system meets the intent of IEEE-279 for nuclear power plant protection systems"
B 3/4 3-5	Add information to the description of source range monitors revise description of radioactive gas effluent monitoring instrumentation

3.4 Reactor Coolant System TSPS (041, 074, 093)

Page	Change
3/4 4-14	Change "REPORTABLE OCCURRENCE" to "Special Report" and add "within 30 days" at end of sentence
B 3/4 4-1	Correct grammar
B 3/4 4-6	Correct typographical errors

3.5 ECCS (TSPS 256, 322)

Page	Change
B 3/4 5-1, B 3/4 5-2	Correct HPCS pump capacity and operating range in Bases

Page	Change
B 3/4 5-2	Change description in Bases of the reactor pressure at which LPCS and LPCI can inject water into the reactor

3.6 Containment Systems (TSPS 003, 004, 019, 062, 093, 107, 128, 144, 164, 167, 170, 171, 172, 240, 266, 269, 276, 311, 320, 379, 818

Page	Change
3/4 6-1	Change "equipment hatch seals" to "seals"
3/4 6-1 3/4 6-13	Change "OPERABLE per" to "is in compliance with the requirements of"
3/4 6-9 3/4 6-17	Change "Specification 4.6.1.6" to "Specification 4.6.1.6.1". Change "reported to the Commission pursuant to Specification 6.9.1" to in a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days"
3/4 6-17	Change "Specification 4.6.2.4" to "Specification 4.6.2.4.1"
3/4 6-27	Add after Isolation Valves "except MSIVs"
3/4 6-32	Change valve names to correspond to plant terminology
3/4 6-46, B 3/4 6-6	Add "rupture discs" to Surveillance Requirement in Secondary Containment Integrity
3/4 6-53	Insert "continuous" before hours
3/4 6-54	Change "high" to "high, high"
3/4 6-56	Delete "and drywell" from specification regarding hydrogen recombiner systems
3/4 6-58	Delete "continued" after the title, Surveillance Requirements
B 3/4 6-1, B 3/4 6-3	Add description of permissible local leak rate test for containment air lock seal
B 3/4 6-2	Change limiting range of building differential pressure from "-2.0 to 0.0" to "-0.1 to 1.0"
B 3/6 6-4	Change reactor blowdown pressure from "1089 psia" to "1060 psia"
B 3/4 6-3	Add description of methods for computing drywell bypass leakage
B 3/4 6-5	Add description of information in Specification Table 3.6.4-1
B 3/4 6-6	Add information regarding surveillance requirements
B 3/4 6-7	Change publication date for Regulatory Guide 1.7 from "March 1971" to "September 1976"

3.7 Plant Systems (TSPS 002, 003, 017, 062, 093, 286, 299, 311)

Page	Change
3/4 7-3, 3/4 7-30	Correct typographical error
3/4 7-4	Add clarification to footnote that if cooling tower fan is running, it does not have to be stopped to test it
3/4 7-5	Insert "continuous" before hours
3/4 7-6	Change isolation valve signal from "low" reactor water level to "low-low" level and from "high" radiation to "high-high" radiation
3/4 7-7	Change "or" to "otherwise" in the Action Statement for the RCIC
3/4 7-27	Add "pursuant to Specification 6.9.2" to the Reports Section
3/4 7-33	Change auxiliary building elevation from 139'6" to 139'0"
3/4 7-34	Change required Co_2 storage tank level from 50% to 60%
3/4 7-43	Delete "in lieu of any report required by Specification 6.9.1"
B 3/4 7-1	Add description on design of SGTS and control room emergency filtration system
B 3/4 7-1	Change operating conditions for LPCS and LPCI

3.8 Electrical Power Systems (TSPS 061, 074, 093, 134, 177, 275)

Page	Change
3/4 8-6	Correct reference to Specification in footnote to 4.8.1.1.2.d.7.a
3/4 8-7	Change reporting requirement to be consistent with other Specifications
3/4 8-13	Add lines to separate rows and columns in table
3/4 8-9	Change minimum water level during refueling from 23 feet to 22 feet 8 inches.
3/4 8-14	Change "with the above" to "with any of the above" and "battery" to "battery bank"
3/4 8-38	Delete "take administrative action to" from an Action Statement

3.9 Refueling Operations (TSPS 182, 275)

Page	Change
3/4 9-1	Change "all rods in" to "all rods out"
3/4 9-10, 3/4 9-16, 3/4 9-17, B 3/4 9-2	Change minimum water level during refueling from "23 feet" to "22 feet 8 inches"

3.10 Special Test Exceptions (TSPS 184)

Page	Change
3/4 10-3	Change "rod-out-notch-override" to "continuous withdrawal"

3.11 Radioactive Effluents (TSPS 087, 093, 105, 193, 248)

Page	Change
3/4 11-5, 3/4 11-6	Delete "in lieu of any other report required by Specification 6.9.1"
3/4 11-7, 3/4 11-12, 3/4 11-13, 3/4 11-14, 3/4 11-15, 3/4 11-16, 3/4 11-18, 3/4 11-20	Delete reference to Section 6.9.1.11
3/4 11-11	Add a sentence to a note to clarify conditions for which the requirements do not apply
3/4 11-15	Change "the site" to "each reactor unit to areas at and beyond the site boundary"
3/4 11-16	Clarify applicability to be "whenever the main condenser offgas treatment system is in operation"
B 3/4 11-4	Correct section number for Sections 3/4 11.2.4, 3/4 11.2.5, and 3/4 11.2.6

3.12 Radiological Environmental Monitoring (TSPS 093)

Page	Change
3/4 12-1	Change reporting requirement to be "pursuant to Specification 6.9.2"
3/4 12-11	Delete "In lieu of a Licensee Event Report and pursuant to Specification 6.9.1.12"

5.0 Design Features (TSPS 074, 105, 225, 258, 281, 282, 283)

Page	Change
5-1	Change "Unrestricted Area Boundary" to "Unrestricted Area and Site Boundary"; capitalize unrestricted area
5-1	Change "Reactor Building" to "Auxiliary Building"
5-3	Correct typographical error
5-3	Make Figure 5.1.2-1 more legible
5-4	Make Figure 5.1.3-1 more legible
5-5	Edit description of fuel assemblies and control rod assemblies and clarify values of enrichment
5-6	Change minimum spent fuel pool level from "202'6"" to "202'5¾"."

6.0 Administrative Controls (TSPS 006, 290)

Page	Change
6-10	Change Specification number referred to from "6.5.2.3" to 6.5.2.2"
6-23	Delete reference to tables listing snubbers

16.5 Summary and Conclusions

The licensee has conducted a comprehensive technical specifications review program to identify and resolve problems with the Grand Gulf Technical Specifications. The program included resolution of problems previously identified by the licensee and by the NRC staff. The comprehensive review included comparison of the Technical Specifications with the as-built equipment, the licensee's Final Safety Analysis Report (FSAR), and the NRC staff's Safety Evaluation Report (SER). The results of the licensee's review program were summarized in Technical Specification Problem Sheets (TSPS), which included a description of the problem and the anticipated resolution.

The NRC staff has reviewed the licensee's comprehensive review program and the TSPS resulting from the program. Based on its review of the program, the staff has concluded that there is reasonable assurance that the licensee's program has accomplished its objectives to identify and make changes to the Technical Specifications as necessary for consistency with the as-built plant configuration, the FSAR, and the SER.

Based on its review of the problems and anticipated resolutions identified in the TSPS resulting from the licensee's program, and based on discussions with the licensee's representatives and staff visits to the plant, the staff concludes that many of the problems do not require changes to the Technical Specifications before exceeding 5% power (see Section 16.3 of this report). The reasons for not requiring changes follow:

(1) A design change is required, and modified equipment is required to be installed and operable before restart after the first refueling outage (see Section 16.3.1(1) of this report).

(2) A design change may result from analyses and tests, and if such a change is found to be necessary, the equipment modifications will require Technical Specification changes at the time they are installed (see Section 16.3.1(2) of this report).

(3) There was not adequate justification for making Technical Specification changes that would be in the direction of reduced safety (see Section 16.3.2 of this report).

(4) The FSAR, rather than the Technical Specifications, should be changed to resolve any inconsistencies. Some of these changes have been made in FSAR Amendment 58 and they were reviewed by the staff and found to be acceptable (see Section 16.3.3(1) of this report). Other changes to the FSAR are required to be made at the next annual update of the FSAR in 1985 (see Section 16.3.3(2) of this report).

(5) The problems were only apparent because further review of the Technical Specifications, plant procedures, design specifications, and the FSAR and SER showed that no problem existed with the Technical Specifications (see Section 16.3.4 of this report).

(6) The TSPS, which were assigned numbers as the problems were identified, were superseded by other problem sheets (see Section 16.3.5 of this report).

The staff met with the licensee during May and June 1984 to discuss those problem areas determined by the staff to require changes to the Technical Specifications before exceeding 5% power. Results of the discussions were recorded on a marked-up copy of the Technical Specifications. The licensee then submitted letters requesting specific changes and providing safety analyses of the changes.

Based on its review of the licensee's proposed changes to the Technical Specifications, the staff has concluded that:

(1) The changes to design, equipment, and associated Technical Specifications identified in Section 16.3.1(1) should be made at the first refueling outage. Three of these design changes were identified during the staff's review of the Technical Specifications: emergency override of the HPCS diesel generator test mode (TSPS 333); a second level of undervoltage protection for the Division 3 power electrical power bus (TSPS 373); and use of coincident logic for diesel generator protective trips, except for engine overspeed and generator differential current (TSPS 808). The license will be conditioned to require these three changes at the first refueling.

(2) The FSAR should be amended by the next annual updating to incorporate the changes committed to by the licensee and identified in Section 16.3.3(2) of this report.

(3) The following changes should be made to the licensee's proposed Technical Specification (TS) changes submitted by letter:

(a) The present surveillance requirement to measure control rod scram accumulator leakage should be retained and not deleted as requested by the licensee (Technical Specification page 3/4 1-9; Section 16.3.2 of this report, TSPS 214).

(b) A proposed surveillance requirement for the source range monitors would require visual inspection that sources are "full in" when the detector drive motor module is removed for maintenance. This surveillance should not be required because when the motor is removed, a control rod block is initiated (Technical Specification pages 3/4 3-53 and 3/4 3-54; Section 16.4.1 of this report, subsection 3.3.6).

(c) The high-pressure alarm setpoints for the low-pressure core spray system (LPCS) and the low-pressure coolant injection (LPCI) system should be 575 psig and 475 psig, respectively, based on system design pressures of 600 psig and 500 psig, respectively. The licensee had proposed 600 psig for the LPCS and 493 psig for the LPCI (Technical Specification page 3/4 5-4; Section 16.4.1 of this report, subsection 3.5(2)).

(d) The Technical Specifications for maintaining an onsite culvert unblocked should be retained, instead of making it a plant procedure as proposed by the licensee. The culvert is designed to pass flood water during a probable maximum precipitation (Technical Specification page 3/4 7-46; Section 16.4.1 of this report, subsection 3.7.10).

(e) The 6-month channel functional test for the reactor protection system electric power monitoring assemblies should be retained. The licensee proposed to perform this test each time the plant is in cold shutdown for a period of more than 24 hours (Technical Specification page 3/4 8-46; Section 16.3.2 of this report, TSPS 181).

(f) A footnote should be added to the proposed minimum source range monitor count rate of 0.7 counts per second required for operability to say "provided signal to noise ratio > 2, otherwise 3 counts per second" (Technical Specification page 3/4 9-4; Section 16.4.1 of this report, subsection 3.9.2(2)).

(g) A proposed exception to the requirement that core alterations be supervised by licensee personnel having a Senior Reactor Operator's license should not be allowed (Technical Specification page 6-1; Section 16.3.2 of this report, TSPS 289).

(h) A licensee proposed change to allow the GGNS Safety Review Committee to delegate its review functions should not be made (Technical Specification page 6-10; Section 16.4.1 of this report, subsection 6.0).

(i) The unit organization chart should be modified to show a dashed line indicating direct access of the Chemistry/Radiation Control Supervisor to the General Manager (Technical Specification page 6-4, Section 16.4.1 of this report, subsection 6.0(1)).

(j) The proposed snubber specification should be modified to make the accelerated inspection schedule, based on the number of inoperable snubbers found during visual surveillance inspections, applicable to all systems rather than just the system on which the inoperable snubber was found (Technical Specification pages 3/4 7-9, 3/4 7-10, 3/4 7-11, 3/4 7-12, B 3/4 7-2, Section 16.4.1 of this report, Subsection 3.7.4)

Amendment 13 to the Technical Specifications incorporates the changes proposed by the licensee, as modified by the above changes. These changes to the licensee's proposals were discussed with the licensee.

The staff's overall conclusion is that with the issuance of Amendment 13 to the Technical Specifications, there is reasonable assurance that Grand Gulf Unit 1 can be operated at full power without endangering the health and safety of the public.

APPENDIX A

CONTINUATION OF CHRONOLOGY

March 14, 1984	Meeting with licensee to discuss requests for changes to Technical Specification. (Summary issued March 20, 1984.)
April 20, 1984	Letter to licensee requesting additional information regarding Technical Specifications problem areas.
April 26, 1984	Meeting with licensee to discuss the means for resolution of Technical Specification problems. (Summary issued May 16, 1984.)
May 11, 1984	Matrices resulting from licensee's Technical Specification Review Program.
May 11, 1984	Licensee markup of Technical Specifications proposed to resolve problem areas.
June 4, 1984	Letter to licensee requesting additional information regarding review personnel for the Technical Specification Review Program.
June 8, 1984	Letter from licensee describing GE Company's overview of the Technical Specification Review Program.
June 9, 1984	Letter from licensee transmitting requests for changes to Technical Specification regarding fuel handling equipment.
June 15, 1984	Letter from licensee requesting changes in correspondence address list.
June 17, 1984	Letter from licensee requesting additional Technical Specification changes related to as-built plant consistency, enchancements that are consistent with the safety analyses, regulatory requirements, requests, recommendations, and correction of typographical errors.
June 17, 1984	Letter from licensee requesting changes to Technical Specifications regarding accident evaluation, containment systems, and materials engineering.
June 18, 1984	Letter from licensee requesting amendment to operating license.
June 18, 1984	Letter from licensee comparing Grand Gulf and Kuo Sheng Technical Specifications.

June 18, 1984	Letter from licensee requesting changes to Technical Specifications regarding fire protection and chemistry.
June 19, 1984	Letter from licensee requesting amendment to operating license.
June 19, 1984	Letter from licensee requesting extension of time period for responding to Jacksonians United for Livable Energy Policies 2.206 petition.
June 19, 1984	Letter from licensee requesting changes to Technical Specifications regarding radiological environmental monitoring.
June 20, 1984	Letter from licensee requesting amendment to operating license.
June 20, 1984	Letter from licensee requesting withdrawal of a proposed change to the Technical Specifications.
June 20, 1984	Letter from licensee requesting changes to Technical Specifications regarding radiological effluents and flood protection.
June 21, 1984	ASLB issues Memorandum and Order (denying licensee's motion for reconsideration or certification).
June 21, 1984	Letter from licensee requesting amendment to operating license.
June 21, 1984	Letter from licensee transmitting requests for changes to Technical Specifications regarding core performance and electrical power systems.
June 22, 1984	Letter to licensee requesting additional information regarding application for exemption to GDC 17, Appendix A to 10 CFR 50, submitted June 4, 1984.
June 22, 1984	Letter from licensee requesting changes to Technical Specifications regarding reactor systems, instrumentation and controls, auxiliary systems, mechanical components, reporting requirements, administrative controls, and operating organization.
June 26, 1984	Letter to licensee concerning delay for FSAR and as-built drawing update.
June 27, 1984	Generic Letter 84-16 issued regarding adequacy of on-shift operating experience for near-term operating license applicants.
June 29, 1984	Letter to licensee transmitting NRC staff markup of Technical Specifications to show resolution of problem areas.
July 2, 1984	Letter from licensee concerning standby diesel generator inspection order, clarification of test requirements.
July 3, 1984	Letter from licensee concerning Technical Specification problem sheet resolutions.

July 3, 1984	Letter from licensee concerning organization and qualifications of management staff.
July 3, 1984	Generic Letter 84-17 issued regarding annual meeting to discuss recent developments regarding operator training, qualifications, and examinations.
July 3, 1984	Letter from licensee providing commitments to resolve three design problems on Grand Gulf electrical power supply.
July 5, 1984	Letter from licensee concerning Technical Specification review program completion.
July 5, 1984	Letter from licensee submitting Transamerica Delaval, Inc. (TDI) diesel generator inspection results.
July 5, 1984	Letter from licensee responding to Region II concerns, including unique features review program.
July 5, 1984	Letter from licensee responding to NRC Region II Report No. 50-416/84-11 regarding deficiencies in Technical Specification Review Program.
July 5, 1984	Letter from licensee providing summary of Grand Gulf Technical Specification Review Program.
July 11, 1984	Letter from licensee concerning organization of management and staff, supplemental information.
July 13, 1984	Letter from licensee concerning Technical Specification Problem Sheet 181.
July 13, 1984	Meeting with licensee to discuss the inspection of Transamerica Delaval, Inc. (TDI), diesel engines.
July 13, 1984	Letter from licensee providing additional information regarding the functioning of the RPS electric power monitoring assembly during surveillance tests.
July 17, 1984	Letter to licensee requesting additional information regarding Transamerica Delaval, Inc. (TDI) engine inspection.
July 19, 1984	Letter from licensee regarding additional information on exemption request for GDC 17.
July 19, 1984	Letter to licensee transmitting final draft of Full-Power License Amendment to Technical Specifications (Amendment No. 13) and requesting revision.
July 20, 1984	PNL report 5201 entitled "Review and Evlauation of TDI Diesel Engine Reliability and Operating - Grand Gulf Nuclear Station Unit 1" submitted through licensee.

July 20, 1984	Letter from licensee requesting changes to the Grand Gulf Technical Specifications regarding surveillance test of diesel generators.
July 25, 1984	Letter from licensee providing additional information regarding channel functional tests of the RPS electric power monitoring assemblies.
July 26, 1984	Letter from licensee concerning gas turbine generator environmental impact information.
July 26, 1984	Letter from licensee supplementing application for partial, temporary exemption to 10 CFR 50, Appendix A, Criterion 17.
July 26, 1984	Letter from licensee providing additional information concerning Transamerica Delaval, Inc. (TDI) engine inspection.
July 28, 1984	Letter from licensee requesting exemption in accordance with 10 CFR 50.12(a) (Division I, II, III Diesel Generators).
July 28, 1984	Letter from licensee concerning Division 1 and 2 Transamerica Delaval, Inc. (TDI) diesel generators.
July 28, 1984	Letter from licensee requesting exemption to 10 CFR 50 Appendix J (Containment Air Lock Testing).
July 30, 1984	Letter from licensee providing supplemental information on the Division 1, TDI Diesel Generator Inspection.
July 30, 1984	Letter from licensee concerning solenoid valves for safety relief valves.
August 2, 1984	Letter from licensee concerning diesel generator turbochargers.
August 2, 1984	Letter from licensee providing supplemental information concerning standby diesel generator combustion air intake and exhaust system.
August 2, 1984	Letter from licensee providing supplemental information concerning request for exemption for Division I, II, III diesel generators.
August 2, 1984	Letter from licensee concerning diesel generator turbochargers.
August 3, 1984	Letter to licensee requesting certification of Technical Specifications.
August 3, 1984	Letter from licensee providing additional information regarding TDI diesel generators.
August 3, 1984	Letter from licensee providing supplemental information regarding standby diesel generator combustion air intake and exhaust system.

August 5, 1984	Letter from licensee certifying Grand Gulf Technical Specifications considering changes in Amendment 13 accurately reflect the plant and safety analyses.
August 5, 1984	Letter from licensee regarding precautionary note on diesel generator loading.
August 5, 1984	Letter from licensee providing supplemental information regarding qualification of solenoid valves for safety relief valves.
August 6, 1984	Letter from licensee responding to staff's request for information regarding senior reactor operator participants in the Technical Specification Review Program.
August 7, 1984	Letter from licensee concerning environmental impact of requested exemptions to 10 CFR 50.
August 13, 1984	Letter from licensee requesting a schedular exemption from 10 CFR 50, Appendix J, and providing partial results of leakage tests of feedwater isolation valves.
August 14, 1984	Letter from licensee requesting additions to Grand Gulf Technical Specifications.
August 30, 1984	Letter from licensee providing additional results of leakage testing of feedwater isolation valves.

APPENDIX B

BIBLIOGRAPHY

General Electric Company, Service Information Letter (SIL) No. 330, June 9, 1980.

---, GESSAR II (NEDO-24011-P-A-6).

Murphy, K. G., and K. M. Campe, "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," in Proceedings of the Thirteenth AEC Air Cleaning Conference, Vol. 1, CONF-740807, March 1975.

U.S. Nuclear Regulatory Commission, NUREG-0123, Revision 3, "BWR Standard Technical Specifications," December 1980.

---, NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," November 1978.

---, NUREG-0473, Revision 3, "Standard Radiological Effluent Technical Specifications for Pressurized Water Reactors," January 1983.

---, NUREG-0700, "Guidelines for Control Room Design Reviews," September 1981.

---, NUREG-0731, "Guidelines for Utility Management and Technical Resources: Draft Report for Interim Use and Comment," September 1980.

---, NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

---, NUREG-0737, Supplement 1, "Clarification of TMI Action Plan, Requirements," December 1982.

---, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," July 1981.

---, NUREG-0831, "Safety Evaluation Report Related to the Operation of Grand Gulf Nuclear Station, Units 1 and 2," September 1981.

---, NUREG/CR-0660, "Enhancement of On-Site Emergency Diesel Generator Reliability," February 1979.

APPENDIX F

NRC REVIEW TEAM

Mr. M. Dean Houston is the NRC Project Manager for this project. Mr. Houston may be contacted at the U.S. Nuclear Regulatory Commission on 301/492-8358.

The principal NRC staff reviewers for this project are:

Name	Title	Branch
R. A. Benedict	Sr. Nuclear Engineer (Management Systems)	Licensee Qualification
W. L. Brooks	Reactor Physicist	Core Performance
S. P. Chan	Sr. Structural Engineer	Structural and Geotechnical Engineering
T. E. Collins	Nuclear Engineer	Reactor Systems
B. J. Elliot	Materials Engineer	Materials Engineering
D. R. Hoffman	Reactor Engineer	Standardization and Special Projects
L. L. Kintner	Senior Project Manager	Licensing Branch 1
J. Y. Lee	Nuclear Engineer	Meteorology and Effluent Treatment
Y. C. Li	Mechanical Engineer	Mechanical Engineering
E. H. Markee, Jr.	Sr. Meteorologist	Meteorology and Effluent Treatment
W. W. Meinke	Health Physicist	Radiological Assessment
E. L. Murphy	Materials Engineer	Operating Reactors Assessment
A. Notafrancesco	Containment Systems Engineer	Containment Systems
J. B. Read	Sr. Physical Scientist	Accident Evaluation
S. C. Rhow	Electrical Engineer (Reactor Systems)	Power Systems
J. N. Ridgely	Mechanical Engineer	Auxiliary Systems

G. B. Staley	Hydraulic Engineer	Environmental and Hydraulic Engineering
J. F. Stang, Jr.	Fire Protection Engineer	Chemical Engineering
M. J. Virgilio	Sr. Reactor Engineer (Instrumentation)	Instrumentation & Control Systems
F. J. Witt	Chemical Engineer	Chemical Engineering

The Pacific Northwest Laboratory (PNL) participated in this review as consultant to the staff. Principal contributors are identified in the PNL report in Appendix M to this report.

APPENDIX M

REVIEW AND EVALUATION OF TRANSAMERICA DELAVAL, INC., DIESEL ENGINE RELIABILITY AND OPERABILITY -- GRAND GULF NUCLEAR STATION UNIT 1

REVIEW AND EVALUATION OF TRANSAMERICA DELAVAL, INC., DIESEL ENGINE RELIABILITY AND OPERABILITY - GRAND GULF NUCLEAR STATION UNIT 1

July 1984

Prepared for
Division of Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
under Contract DE-AC06-76RLO 1830
NRC FIN B2963

Project Title: Assessment of Diesel Engine Reliability/Operability

NRC Lead Engineer: C. H. Berlinger

Pacific Northwest Laboratory
Richland, Washington 99352

PACIFIC NORTHWEST LABORATORY

PROJECT APPROVALS

W. W. Laity

W. W. Laity, Project Manager
Pacific Northwest Laboratory

Date July 20, 1984

W D Richmond

W. D. Pichmond, Chairman
Senior Review Panel
Pacific Northwest Laboratory

Date 7-20-84

CONTENTS

REVIEW AND EVALUATION OF
TRANSAMERICA DELAVAL, INC., DIESEL ENGINE
RELIABILITY AND OPERABILITY - GRAND GULF NUCLEAR STATION UNIT 1

1.0 INTRODUCTION

In support of its request for a full power license of Grand Gulf Nuclear Station (GGNS) Unit 1 and in response to an NRC Order dated May 22, 1984, Mississippi Power & Light Company (MP&L) submitted a report on July 5, 1984, addressing three areas:

- a description of the June 1984 disassembly and inspection of the Division I diesel generator
- the post-inspection engine test program
- proposed enhancements to the MP&L maintenance and surveillance program.

As also required by the NRC Order, the MP&L submittal addresses the similarity of the "as-manufactured quality" of the Division I and II diesel generators as part of MP&L's justification for not inspecting the Division II engine. These diesel generators are Model DSRV-16-4 manufactured by Transamerica Delaval, Inc. (TDI).

This Technical Evaluation Report (TER) documents Pacific Northwest Laboratory's (PNL) evaluation of the reliability and operability of the Division I and II diesel generators at GGNS Unit 1. In addition to the July 5, 1984 submittal, PNL has reviewed MP&L submittals dated February 20, April 17, and May 6, 1984. Other information, identified herein, was also considered as needed to support conclusions.

The TER organization is as follows: Section 2 provides background on the TDI problem resolution by both the group of nuclear utility TDI owners and MP&L. Section 3 provides a detailed review and evaluation of the Division I engine disassembly and inspection. Section 4 reviews the MP&L report on the comparability of the Division I and Division II engines. Sections 5 and 6 document PNL's review/evaluation of MP&L's post-inspection engine tests and the utility's proposed augmented maintenance/surveillance program, respectively. Finally, Section 7 presents PNL's overall conclusions and recommendations regarding the two engines' suitability to serve as standby power sources at the GGNS.

This TER was prepared by the following PNL staff and consultants:

- D. A. Dingee, PNL project staff
- A. J. Henriksen, diesel consultant to PNL
- J. E. Horner, representing Seaworthy Systems, Inc., diesel consultants to PNL
- P. J. Louzecky, Engineered Applications Corporation, diesel consultant to PNL.

Others whose contributions were considered in formulating the conclusions include PNL Assesment of Diesel Engine Reliability/Operability Project Team members J. M. Alzheimer, M. Clement, S. D. Dahlgren, R. E. Dodge, W. W. Laity, J. F. Nesbitt, J. C. Spanner, and F. R. Zaloudek; and consultants S. H. Bush, B. J. Kirkwood (Covenant Engineering), and J. A. Webber (representing Ricardo Engineering).

2.0 BACKGROUND

2.1 OWNERS' GROUP PROGRAM PLAN

Thirteen nuclear utilities that own diesel generators manufactured by Transamerica Delaval, Inc. (TDI), have established an Owners' Group to address questions raised by a major failure in one TDI diesel (at the Shoreham Nuclear Power Station in August 1983), and other problems in TDI diesels reported in the nuclear and non-nuclear industry. On March 2, 1984, the Owners' Group submitted a plan to the U.S. Nuclear Regulatory Commission (NRC) outlining a comprehensive program including 1) an in-depth assessment of 16 known engine problems (Phase I), 2) a design review and quality revalidation program that addresses other key engine components (Phase II), and 3) engine tests and inspections. A review of that submittal was conducted by PNL and reported to NRC in PNL-5161 dated June 1984.

Section 4 of PNL-5161 deals with considerations for interim licensing of nuclear stations prior to completion of the implementation of the Owner's Group Program Plan. Recommendations relevant to MP&L licensing of the GGNS at this time are:

- The engine should have AE pistons or complete "lead-engine" tests as described in Section 2.3.2 of PNL-5161.
- The diesel generator should not be required to carry a load in excess of that corresponding to engine Brake Mean Effective Pressure (BMEP) of 185 psig.
- The engine should be inspected per Section 2.3.2.1 of PNL-5161 to confirm that the components are sound.
- Pre-operational testing should be performed as discussed in Section 2.3.2 of PNL-5161.
- The engines should receive enhanced surveillance and maintenance.

2.2 GRAND GULF NUCLEAR STATION

An MP&L submittal to NRC, dated February 20, 1984, provided a review of the results of their program of inspection, upgrading, testing and maintenance. The PNL review of this document was provided to NRC in a letter dated March 30, 1984. A number of concerns were identified by PNL, namely:

- The MP&L report did not provide sufficient information to convince the reviewers that the AE pistons were suitable for GGNS licensing.
- The evidence was insufficient to conclude that the cylinder heads would perform reliably.
- The connecting rod bearings were not demonstrated to be suitable for operation at GGNS.
- The push rods were not adequately tested.
- Data concerning crankshaft deflections and main bearing wear were needed to confirm the adequacy of the crankshaft.
- The high-pressure fuel line needed to be examined to assure the reviewers that the new lines installed at GGNS are not defective.
- MP&L did not adequately consider the possibility of cracks in the cylinder block.
- Additional information was needed to confirm that the engine base would not crack.
- MP&L did not address head stud problems noted by the Owners' Group.
- The issues on rocker arm capscrews were not closed out per the Owners' Group recommendations.
- The PNL reviewers needed more information from MP&L on turbocharger mounting.
- The evidence provided by MP&L on the connecting rods was insufficient to conclude that they would be adequate.

- MP&L did not address the potential for wrist pin bushing failures; PNL noted that cracks had been observed in wrist pin bushings at the Shoreham Nuclear Power Station.
- The test program was deemed to be inadequate.
- The description of the surveillance and maintenance program was insufficient for the PNL reviewers to draw conclusions.

This detailed PNL review of the February 20 MP&L submittal was followed by a letter dated April 16, 1984, in which PNL recommended a number of actions to support licensing of the GGNS. These included 1) inspection of one engine at GGNS, 2) post-inspection testing, and 3) maintenance and surveillance items. In a letter dated April 17, 1984, PNL provided additional clarification on these actions.

On April 25, 1984, NRC issued a letter to MP&L identifying these actions as an acceptable basis to support full power operation at GGNS for one fuel cycle pending completion of the Owners' Group Program Plan.

After considering additional, updated information provided by MP&L by letter dated May 6, 1984, NRC issued an Order dated May 22, 1984, requiring disassembly and inspection of one engine before the power ascension program could be authorized. Comments pertaining to the need for these inspections were provided in a PNL letter dated May 21, 1984.

On June 4 and 5, 1984, PNL staff and consultants visited MP&L to review the Division I engine components. A PNL letter dated July 9, 1984, summarized the results of this inspection. In general, the inspection did not reveal any problems that should seriously impact the reliability and operability of the engine for the first reactor fuel cycle.

On July 5, 1984, MP&L provided NRC with a report on the Division I disassembly and inspection, in response to the May 22, 1984, Order. This report also compared the Division II diesel generator (DG) to the Division I DG, and addressed post-inspection testing and a proposed augmented maintenance and surveillance program aimed at assuring the future satisfactory performance of both engines. This submittal was the topic of discussion at a meeting held July 13, 1984, among representatives of MP&L, NRC, and PNL.

3.0 EVALUATION OF MP&L DIVISION I ENGINE DISASSEMBLY AND INSPECTION

In compliance with the NRC Order of May 22, 1984, MP&L disassembled the Division I TDI engine and inspected all critical components. These components included those that are being addressed as part of the Owners' Group Phase 1 Program regarding known generic problem areas: cylinder heads, engine block and base, connecting rods, pistons, studs, cap screws, push rods, etc. The specific inspection methods used were identified in the NRC Order. Actions taken by MP&L in conducting the disassembly and inspection are consistent with Section 2.3.2.1 of PNL-5161 dealing with pretest inspections.

This section documents PNL's technical evaluation of MP&L's resolution of each of the 16 known generic problems (components) as well as 8 problems specific to GGNS. It consists of worksheets providing 1) component identification, 2) a brief history of failures, 3) the status of the Owners' Group Program aimed at resolving the problem, 4) the status of MP&L in resolving the problem, and 5) PNL comments/conclusions. PNL's conclusions and comments are based not only on the MP&L submittal of July 5, 1984, and the related discussions on July 13, 1984, but also on an onsite inspection of the engine components. It must be emphasized that, pending completion of the implementation of the Owners' Group Program Plan, PNL's conclusions are plant-specific, applying only to MP&L's Grand Gulf Nuclear Station Unit 1 and to operations only during its first reactor refueling cycle. It is understood that, at the first refueling, MP&L will implement all applicable recommendations of the Owners' Group.

The order of worksheet presentation is as follows. The 16 known problems are reviewed in the order listed in PNL-5161, Table 1. Next, the GGNS-specific problems are reviewed in the following order: low-presure fuel lines, crankcase cover capscrews, fuel oil leaks, air start valve failures, air start solenoid valve failure, fuel oil injection pump, cracks in air box, and failures to start Division I engine.

3.1 GENERIC PROBLEMS

Component: Piston Skirt
Part No. 03-341-04-AE
Owners' Group Report: FaAA-84-2-14

Brief History of Failures

Based on a number of cracks found in AF piston skirts at GGNS, Shoreham, and at non-nuclear installations, the skirt design was strengthened in the boss area where the cracks had been found. No failures have been reported to date on the redesigned piston skirt, labeled AE, in either nuclear or non-nuclear installations. Kodiak has operated in excess of 6000 hours at approximately 185 BMEP (1200 psi maximum pressure); the TDI R-5 test engine in excess of 600 hours with maximum pressures of 2000 psi.

Owners' Group Status

The Owners' Group consultant, Failure Analysis Associates (FaAA), has analyzed the AE piston skirt design and has concluded that the AE skirts may crack at 10% overload, but that cracks will not propagate to the point of failure.

MP&L Status

After observing cracks in several skirts, all AF skirts on both Division I and II engines were replaced with AE skirts in January/February 1984. Subsequently, after 270 hours of operation, all Division I skirts were inspected by liquid penetrant and no rejectable indications were observed. However, the piston skirt-to-crown surface on all skirts and crowns showed slight signs of fretting due to relative movement.

PNL Conclusions

PNL has reviewed both the Owners' Group report and the relevant inspection data. Based on this review, as well as on the aforementioned operating experience with the Kodiak and R-5 engines, PNL concludes that the piston skirts are acceptable for operation up to and including 185 BMEP at 450 rpm (the 185 BMEP criterion is discussed in PNL-5161, Section 4, "Considerations for Interim Licensing").

Component: Connecting Rod Bearing Shell

Part No. 02-340-04-AG

Owners' Group Report: FaAA-84-31

Brief History of Failures

No failures of the V-engine connecting rod (conrod) bearing shells have been reported in nuclear applications. However, a number of bearings have been replaced due to nonconformity with Owners' Group recommendations.

Owners' Group Status

Failure Analysis Associates has conducted both stress and orbital analyses of the conrod bearing shells. Provided the shells are dimensionally correct and otherwise conform to specifications as recommended by the FaAA report, FaAA has concluded that the bearings are suitable for the service intended.

MP&L Status

In January/February 1984 all conrod bearings in both engines were replaced as a matter of policy. In June 1984, after 270 hours of operation, the Division I engine shells were inspected visually and by liquid penetrant. All bearings (except No. 7) were x-rayed. Bearing No. 7 was sent to FaAA to aid in the ongoing generic analysis. All other bearings were found acceptable in accordance with Owners' Group acceptance criteria. However, bearing No. 4 was replaced nonetheless, due to a 1/2-inch wide wipe caused by dirt. Bearing No. 7 was also replaced; all other bearings were reinstalled.

PNL Conclusions

PNL has reviewed the Owners' Group report and the relevant inspection data, and has visually inspected the bearings. PNL concludes that the bearings are acceptable for the first refueling cycle.

Component: Rocker Arm Capscrew

Part No. 02-390-01-0G

Owners' Group Report: Stone & Webster, March 1984

Brief History of Failures

Rocker arm capscrew failures at Shoreham have been reported. There have been no reports of similar failures elsewhere.

Owners' Group Submittal

Stone & Webster Engineering Corporation, a consultant to the Owners' Group, has performed stress analyses of both the original capscrew design (the type that failed at Shoreham) and a newer design. Stone & Webster has concluded that both designs are adequate for the service intended. Stone & Webster has attributed the failure at Shoreham to undertorquing.

MP&L Status

The rocker arm capscrews at GGNS are of the original design. These capscrews have experienced in excess of 10^7 loading cycles without reported failures. Breakaway torques measured during the June 1984 inspection were within acceptable limits. Torque was checked on all capscrews after reassembly in June 1984.

PNL Conclusions

Based on the analytical results and operating experience to date, PNL concludes that adequate torquing ensures that the capscrews will provide acceptable service.

Component: Air Start Valve Capscrews

Part No. Gb-032-114

Owners' Group Report: Stone & Webster, March 1984

Brief History of Failures

No actual failures of capscrews have been reported. However, on May 13, 1984, TDI reported a potential defect due to the possibility of the 3/4-10 x 3-inch capscrews bottoming out in the holes in the cylinder heads, resulting in insufficient clamping of the air start valves.

Owners' Group Status

Stone & Webster and TDI both have recommended that the 3-inch capscrews be either shortened by 1/4 inch or replaced with 2-3/4-inch capscrews.

MP&L Status

Capscrews on both Division I and II DGs have been modified by shortening the 3-inch capscrews by 1/4 inch. Proper torque values were confirmed after reassembly.

PNL Conclusions

After reviewing available reports and inspection data, PNL concludes that proper corrective measures were taken and that capscrews are acceptable for the first refueling cycle.

Component: Push Rods

Part No. 02-390-06-AB

Owners' Group Report: FaAA 84-3-17

Brief History of Failures

The push rods originally had tubular steel bodies fitted with hardened steel end pieces attached with plug welds. Reportedly, an estimated 2% developed cracks in or around the plug welds. A push rod design introduced later consisted of a tubular steel body with a carbon steel ball fillet welded to each end. This design proved to be very prone to cracking at the weld. In all, 15 of 16 rods on the GGNS Division I engine and 13 of 16 rods on the Division II engine were found to be cracked. All push rods on both Division I and Division II DGs have been replaced by a new design consisting of a tubular steel body with a steel cylinder friction-welded to each end. No failures are reported on this design.

Owners' Group Status

Failure Analysis Associates has a performed stress analysis as well as cycle wear test to 10^7 cycles on a sample of the friction-welded push rod at conditions simulating full engine nameplate loading. No sign of abnormal wear or deterioration of the welded joints was observed.

MP&L Status

All push rods on both Division I and II engines were replaced in January/February 1984 by a new design consisting of a tubular steel body with a steel cylinder friction-welded to each end. During the June 1984 inspection, all push rods were inspected by liquid penetrant and no relevant indications were observed.

PNL Conclusions

After reviewing the FaAA report and inspection data and noting the GGNS replacements, PNL concludes that the push rods incorporating the friction weld design are acceptable for the first refueling cycle.

Component: Cylinder Head Stud

Part No. 03-315-01-0A (Old Design)

Owners' Group Report: Stone & Webster, March 1984

Brief History of Failure

To date, no failure of cylinder head studs has been reported in the nuclear industry. However, some isolated failures have been reported in the non-nuclear field. The cause has not been reported.

Owners' Group Status

Stone & Webster Engineering Corporation has analyzed both the old design studs and the new necked down studs developed by TDI to minimize cylinder block cracking, and has concluded that both stud designs are adequate for the service intended, provided proper stud preload is applied.

MP&L Status

The MP&L visual inspection revealed many instances of flat crests on the top threads of the studs and one instance of minor thread damage to the bottom threads. On the engine left bank cylinder No. 3, studs No. 4 and 5 had a 360° discernable surface indication on the stud shank. None of the thread damage was considered service-related and it was concluded that the damage to stud No. 4 and 5 shanks was done during machining of the studs. These two studs were replaced by new studs. It is believed the replacement studs are of the new necked down design. This will be confirmed by MP&L. The damaged stud threads were chased with a die, re-examined, accepted, and reinstalled. Preload was checked on all studs after installation.

PNL Conclusions

Based on a review of the Owners' Group report and the inspection data supplied in the July 5 submittal of MP&L, PNL concludes that the cylinder head studs are acceptable for the first refueling cycle.

Component: High-Pressure Fuel Tubing

Part No.: 03-365C

Owners' Group Report: Stone & Webster, April 1984

Brief History of Failures

High-pressure (HP) fuel tubing developed leaks during preoperational testing on both the Shoreham and Grand Gulf engines. There are no other reported failures in nuclear applications.

Owners' Group Status

Stone & Webster has analyzed the failed HP fuel tubing and has concluded that the failures originated in inner surface flaws that were initiated during fabrication. If, through eddy current inspection, the inner surface condition of new tubing is found to be within specified conditions, the HP tubing is considered suitable for the service intended.

MP&L Status

Fifteen HP fuel lines on both Division I and II engines are original equipment and have experienced over 10 million operating cycles. Operating stresses are therefore believed to be smaller than the high-cycle fatigue endurance limit, and thus these tubes are believed to be free of detrimental defects to the inner surface. Both replacement tubes, one on each Division engine, have been subjected to the prescribed surveillance and were found to be sound.

PNL Conclusions

PNL has determined that the original high-pressure lines are acceptable, based on their completing 10^7 operating cycles. PNL has also determined that the replacement tubes have been adequately inspected. Thus, PNL concludes that the HP fuel tubing is acceptable for the first refueling cycle.

Component: Crankshaft

Part No. 02-310A

Owners' Group Report: FaAA-84-4-16, (dated May 22, 1984)

Brief History of Failures

Three V-16 crankshaft failures have been reported, all in the non-nuclear industry. Two failures were attributed to torsional stress due to operation too close to the critical speed. No cause has been suggested for the third failure.

Owners' Group Status

Failure Analysis Associates has performed torsional and bending stress analyses of the subject crankshaft and has concluded that the shaft will meet Diesel Engine Manufacturers Association (DEMA) standards at the nameplate rated load and speed. The radius of the fillets in main journal oil holes was identified as an area of potential stress concentration and careful inspection of this area was prescribed.

MP&L Status

At MP&L's request, Bechtel Corporation reviewed the FaAA analysis and conducted an independent dynamic analysis of the crankshaft. Bechtel concluded that the shaft will meet DEMA standards. Torsiograph tests will be conducted to compare operating values with analytical values. During inspections in June 1984, crank fillets were inspected by liquid penetrant and found to be sound. Further, oil hole fillets on main journals No. 4, 6, and 8 were inspected by liquid penetrant with no indications noted. Minor scratches were noted on several crank journals. Also, on crank journal No. 4 a slight metal buildup (rodbearing replaced) was noted; it was removed and the journal polished. Hot and cold crankshaft deflections have been measured and documented and reported to be within TDI and Owners' Group specifications.

PNL Conclusions

Based upon the status of PNL's review of the Owners' Group report prepared by FaAA regarding the crankshaft, PNL is not prepared to agree with the FaAA analysis at this time, and has requested further analytical data from the Owners' Group. PNL has also requested that torsiograph tests be conducted at 0%, 25%, 50%, 75%, and 100% rated nameplate loads and rpm. PNL views the torsiograph data as confirmatory to the analysis. PNL concurs that documented hot and cold crankshaft deflections are within TDI and Owners' Group specifications. On this basis, PNL agrees that the crankshaft will be adequate for operation at loads up to and including 185 BMEP and 450 rpm (as described in PNL-5161 Section 4, "Considerations for Interim Licensing").

Component: Turbochargers

Part No.: Elliott 90G

Owners' Group Report: FaAA-84-5-7

Brief History of Failures

Reports of turbocharger thrust bearing problems are limited to the nuclear industry. To date, thrust bearing problems have been reported for San Onofre, Catawba, and Comanche Peak. Nozzle vane and capscrew problems have also been reported; such problems have occurred at GGNS. Misalignment problems resulting in sheared foundation bolts, as well as broken lube oil return lines and mounting welds, have also been experienced at various nuclear power stations.

Owners' Group Status

In Report No. FaAA-84-5-7, dated May 1984, Failure Analysis Associates has analyzed the turbocharger thrust bearing problems for the model 90G turbocharger and has concluded that the problems are due to insufficient lubrication of the thrust bearings during "fast" starts (i.e., automatic starts for which no prelubrication is provided to the thrust bearing). Several types of startup lubrication systems have been implemented at nuclear power plants to avoid these problems. One type is a drip system that provides lubrication from the before-and-after (B&A) recirculation system. An alternate type (in use at GGNS) is an auxiliary B&A lube oil pump. This pump is activated prior to any planned start and provides the turbocharger bearings with sufficient lube oil to complete fast starts as required for nuclear standby tests.

FaAA states in the above-mentioned report that findings related to nozzle-vane life and nozzle-ring capscrew design will be presented in a following report. Misalignment problems are not addressed in the FaAA report, and are not mentioned as a topic for a following report.

MP&L Status

During the June 1984 inspection of the Division I engine, it was discovered that two nozzle ring capscrew heads and one nozzle ring blade were missing on the right-bank turbocharger. It was assumed that the capscrew heads had passed through the turbine. On the left-bank turbocharger, one nozzle ring capscrew head had broken off, but was still attached to the locking wire. One nozzle ring blade was also found to be missing. Subsequent inspection of the Division II turbocharger revealed one nozzle ring blade missing on each turbocharger. No broken capscrews were found on Division II turbochargers.

MP&L concluded that missing nozzle ring blades had been removed on purpose. The broken capscrews were metallurgically examined and the failure mechanism determined to be intergranular stress corrosion cracking, believed to have been initiated by sulfurous compounds in the exhaust gases during shop tests at TDI. An engineering study by MP&L to determine the need for a different capscrew material is underway.

Division I turbochargers were sent to Elliot for refurbishment, where the thrust bearings, although still serviceable, were replaced. Nozzle ring blades to replace those missing were also installed on both Division I turbochargers.

MP&L has taken extensive actions to correct vibration problems and is confident that earlier misalignment problems resulting in sheared foundation bolts, as well as broken lube oil return lines and mounting welds, are solved through proper alignment.

PNL Conclusions

On the basis of information presented in the FaAA report referenced above, the transcript of the meeting among representatives of FaAA, the Owners' Group, NRC, and PNL on June 22, 1984, and the inspection data presented by MP&L, PNL concludes that the action taken at GGNS to provide lubrication to turbocharger bearings is adequate for the first refueling cycle. Key considerations in support of this conclusion are as follows:

- According to Failure Analysis Associates, as confirmed in a telephone conversation between PNL (W. Laity) and FaAA (T. Thomas) on July 20, 1984, the shortest known time-to-failure of a turbocharger thrust bearing subjected to "dry" starts (for which no bearing prelubrication was provided) occurred at the Shoreham Nuclear Power Station. That bearing experienced at least 62 "dry" starts before failure.

- On the basis of operating experience at GGNS over a 2-year period, MP&L estimates that the diesels may experience two "dry" starts per diesel per year. Turbocharger thrust bearings examined from the Division I engine after two "dry" starts showed no evidence of distress. Float measurements of thrust bearings in the Division II engine are well within manufacturer's specifications, also indicating no thrust bearing distress.

PNL has also reviewed the MP&L actions regarding turbocharger realignment and notes that in excess of 100 hours of operation have occurred without incidents attributable to misalignment or vibration. PNL concludes that MP&L has taken appropriate actions to correct misalignment problems.

In addition, PNL has reviewed the MP&L conclusion that service-related conditions are not responsible for the missing nozzle ring blades. The fact that one blade is missing from each of four nozzle rings (both engines) and that there is a high probability of damage to the turbocharger if the vane breaks in service (not seen on inspection) supports the MP&L conclusion.

On the basis of the above-mentioned analyses, inspections and reviews, PNL concludes that the turbochargers are acceptable for the first refueling cycle.

Component: Connecting Rod

Part No.: 03-340A

Owners' Group Report: FaAA-84-3-14

Brief History of Failures

Connecting rod failures have been reported from the non-nuclear field. Two failure modes have been observed. The first mode was link rod bolt failure due to loss of bolt preload. The second mode of failure was fatigue cracking of connecting rod bolts and/or the link rod box in the mating threads. No connecting rod failures have occurred in nuclear service.

Owners' Group Status

The first failure mechanism is fatigue failure of the link rod bolts resulting from loss of bolt preload. The problem and its solution were addressed by TDI in Service Information Memo No. 349, dated September 18, 1980 (pp. 1-3). According to this SIM, engines manufactured between 1972 and February 1980 may have been shipped with an insufficient locating dowel counterbore depth in the link rod or link pin, resulting in clearance between the link rod and link pin as assembled. Under firing load, this locating dowel will yield, allowing the above clearance to disappear and resulting in loose link rod bolts. The Owners' Group (through the above-mentioned FaAA report) has determined that there must be zero clearance under the specified bolt torque of 1050 ft-lb, and they recommend that the utilities check the clearance with a 0.0015-in feeler gage.

The second failure mechanism is fatigue cracking of the connecting rod bolts and/or the link rod box in the mating threads. TDI attributed these rod cracks to "thread fretting". This "thread fretting" was concluded by TDI to result from distortion of the rod bolt under operating loads in the area of the mating threads; the distortion could occur if the bolts had been installed with the originally specified bolt preloads. The Owners' Group addresses this concern for the two versions of the connecting rod, namely the original design, equipped with 1-7/8-inch bolts and a later design in which the rod boxes are equipped with a 1-1/2-inch bolts. Stress analysis, including finite element, has been completed by FaAA. Failure Analysis Associates has concluded that both designs are adequate for the service intended, provided conrod bolt preload is checked within time limits specified as related to engine load requirement in terms of percentage of nameplate rating. However, the rod with the 1-1/2-inch bolts has an 8% to 9% higher margin of safety than the rod with 1-7/8-inch bolts because the rod box structure is more massive with the smaller bolt configuration.

MP&L Status

With regard to the link rod/link pin clearance, MP&L has performed the Owners' Group recommended measurements described above.

The status of the fatigue cracking in the rod boxes is as follows. Both Division I and II conrods are equipped with 1-7/8-inch conrod bolts. During the June 1984 inspection, all connecting rods and accessory equipment were inspected; the findings and dispositions are as follows:

- Serrated joint teeth surfaces were found to have minor fretting on all conrod boxes. At NRC's request, the serrated teeth were dressed via stoning and the contact surface verified by "blueing" as per TDI specifications.
- Conrod external machined surfaces were inspected by MP&L and revealed no indications.
- Magnified borescopic inspection of female threads indicated pitting in one hole of No. 1, galling in one hole of No. 6, and heavy galling in one hole of No. 5. All conditions were judged to be maintenance- rather than service-induced. Rod No. 5 was replaced and threads in the other rods were tapped and reinspected.
- Conrod bolt inspection revealed that approximately 50% of the bolts had minor galling, which was judged to be maintenance-related. All bolts were replaced with fully inspected new bolts. When bolts were installed, they were properly lubricated as per instructions. Proper preload was ascertained by ultrasonic methods.

- All conrod dimensions were checked and found to be within specified tolerances. All wrist pin bushings were inspected by liquid penetrant and found to be in good condition. MP&L has proposed to check conrod bolt preload after 270 hours operation or at first refueling, whichever comes first.

PNL Conclusions

PNL concurs with the MP&L resolution of the connecting rod problem resulting from link rod/link pin clearance, namely feeler-gage confirmation that no clearance exists.

Relative to the fatigue cracking in the rod bolts and/or the link rod box, PNL has reviewed all available information on the subject, and concludes that, provided the check on conrod bolt preload is carried out after 200 hours of operation or after 9 months, whichever comes first, the conrods are acceptable for the first refueling cycle.

Component: Engine Base and Bearing Cap

Part No.: 03-305C, CSG Class A

Owners' Group Report: FaAA-84-6-53

Brief History of Failures

The only failure reported by the Owners' Group for DSRV-16 engines occurred in a non-nuclear application: a nut pocket failed on a DSRV-16 engine at the ANAMAX mine near Tucson, Arizona. According to FaAA, the engine manufacturer (TDI) reported that this failure was due to impurities in the casting material that reduced the engine base strength.

Owners' Group Status

Failure Analysis Associates has analyzed the base, bearing saddles, bearing caps, nut pockets, and bolting/nuts. FaAA has concluded that the base assembly components have the strength necessary to operate at full rated load for indefinite periods, provided that all components meet their specifications, that they have not been damaged, and that proper preloads are maintained.

MP&L Status

During the June 1984 inspection liquid penetrant techniques were used on the main bearing cap-to-engine base saddle surfaces on main bearings No. 4, 6, and 8. No relevant indications were observed.

PNL Conclusions

Based upon PNL's review of the Owners' Group report and the engine inspection findings reported by MP&L, PNL concludes that the engine base assembly is acceptable for the first refueling cycle.

Component: Cylinder Head

Part No.: 03-360A

Owners' Group Report: FaAA-84-15-12

Brief History of Failures

Numerous reports on cylinder head failures are available from both the nuclear and non-nuclear industry. For identification purposes, TDI cylinder heads are classified as I, II, and III, all under the same part number. Group I are heads cast prior to October 1978; Group II are heads cast between October 1978 and September 1980; and Group III are heads cast after September 1980. Most instances of cracked heads have involved Group I. Only five instances of water leaks in Group II and III heads have been reported, all in marine applications. Many of the cracks initiated at the stellite valve seats.

Owners' Group Status

Failure Analysis Associates mechanical and thermal stress calculations, which did not include finite element calculations, concluded that Group I, II, and III heads as designed are adequate for the service intended. The report recommends that Group I and II heads be inspected by liquid penetrant and magnetic particle as well as ultrasonic testing to determine firedeck thickness. For Group III heads, sample inspection as described above is recommended. For all three groups of heads, barring over before startup is recommended.

MP&L Status

During the June 1984 inspection, all heads (all of which are believed to be Group I) on Division I engines were inspected in accordance with Owners' Group recommendations. Eleven heads met all Owners' Group acceptance criteria. Five heads needed further engineering evaluation before being accepted. MP&L proposed to bar the engine over 4 hours after engine shutdown, and once weekly thereafter. Routinely, the engine will be rolled over prior to a planned start.

PNL Conclusions

PNL has reviewed all the pertinent material and also notes that MP&L will limit the engine load during the first refueling cycle to that corresponding to 185 BMEP. On these bases, PNL concludes that the cylinder heads are acceptable for the first refueling cycle, provided that the engine is rolled over 4 hours after shutdown, 24 hours after shutdown, and thereafter prior to each planned start, to check for water leakage into the cylinders.

Component: Jacket Water Pump

Part No.: 03-425

Owners' Group Report: Stone & Webster, June 1984

Brief History of Failures

Shoreham has experienced a jacket water pump shaft failure on the TDI R-4 engine. There is no history of failures on jacket water pumps designed for the V-16 engines.

Owners' Group Status

Stone & Webster has investigated this design jacket water pump and has concluded that, provided proper care is taken to ensure minimum and maximum torque when installing the nut holding the external spine in the taper, the jacket water pump is adequate for the service intended.

MP&L Status

No problems have been experienced.

PNL Conclusions

Based upon the absence of adverse experience with water pumps designed for the V-16 engines, as well as on the review of the Stone & Webster report, PNL concludes that the jacket water pump is acceptable for the first refueling cycle.

Component: Engine Mounted Electrical Cable

Part No.: 03-688B

Owners' Group Report: Stone & Webster, June 1984

Brief History of Failures

No failure of this part has been reported. However, in TDI Service Information Memo No. 361, TDI reported that three engine mounted cables associated with 1) the Woodward governor/actuator, 2) the Air-Pax magnetic pick-up, and 3) the Air-Pax tachometer relay, represent potential fire hazards.

Owners' Group Status

Stone & Webster carried out a field survey. Based on the survey results, Stone & Webster concluded that Class 1E IEEE 383-1974 qualified cable, as now installed in both the Division I and II engines, meets the intended function and is acceptable for the required operation.

MP&L Status

The original commercial grade cable has been replaced by Class 1E IEEE 383-1974 qualified cable in both Division I and II engines.

PNL Conclusions

PNL concludes that the Class 1E IEEE 383-1974 qualified cable as installed is acceptable for the first refueling cycle.

Component: Cylinder Block

Part No.: 03-315A

Owners' Group Report: FaAA-84-5-4

Brief History of Failures

Numerous incidents of cylinder block failures have been reported in the non-nuclear field. In the nuclear field, all three engines at Shoreham have cracks in their cylinder blocks. At Comanche Peak, cracks were observed after 90 hours of operation.

Owners' Group Status

Failure Analysis Associates performed strain gauge testing combined with two-dimensional analytical modeling of the block top and liner. Based on these efforts, FaAA concluded:

- Eventually, depending upon load and operating hours, cracks will initiate between stud hole and line counterbore. Cracks are predicted to be benign.
- Cracks between stud hole and liner counterbore will increase likelihood of cracks developing between stud holes of adjacent cylinders. The deepest crack measured in this region (5-1/2 inches in depth at Shoreham) did not degrade engine operation or loosen studs.
- Provided there are no cracks between stud holes between adjacent cylinders, the block is predicted to have sufficient margin to withstand a LOOP/LOCA event.

The FaAA report recommends inspections of cylinder blocks at intervals related to load and operating hours.

MP&L Status

At the June 1984 inspection, the Division I cylinder block was inspected in all critical areas by liquid penetrant as recommended by the Owners' Group. No critical indications were observed.

PNL Conclusions

After reviewing the FaAA report, and noting that MP&L found no significant indications on the cylinder block and, further, that MP&L will limit the engine load to that correspondiong to 185 BMEP, PNL concludes that the cylinder block is acceptable for the first refueling cycle, subject to the periodic surveillance proposed by MP&L in Section 6.2 of their July 5 submittal.

Component: Cylinder Liner

Part No.: 02-315-02-0G

Owners' Group Report: FaAA 84-5-4

Brief History of Failures

Only one incident of cylinder liner failure is available. This failure occurred in 1982 at Grand Gulf when a piston crown separated from the skirt during testing of the Division II engine.

Owners' Group Status

The Owners' Group has identified incorrect cylinder liner dimensions as being a contributing factor in liner stresses.

MP&L Status

During the June 1984 inspection all liners were inspected, deglazcd, and reinstalled. Dimensional inspections of the liners were performed by MP&L to ensure that the clamping force of the cylinder head on the liner would not induce excessive stress on the cylinder block.

PNL Conclusions

Based upon the MP&L inspection and determination of correct dimensions, as well as upon PNL's onsite inspection during the June 1984 plant visit, PNL concludes that the liners are acceptable for the first refueling cycle.

3.2 PLANT-SPECIFIC PROBLEMS

Component: Low-Pressure Fuel Lines

Brief History of Failures

On September 4, 1984, the Division I engine was stopped due to a fire that broke out at the engine. The fire was caused by a break in a 1-inch fuel oil supply header. MP&L investigated the failure and concluded that it was due to the absence of a clamp, resulting in excessive vibration.

MP&L Status

MP&L designed and installed a tubing support for this section of tubing on both Division I and II engines. Vibration tests indicated vibration levels to be well within normal levels for this type of machinery.

PNL Conclusions

PNL has reviewed the pertinent MP&L report and determined that the cause of the failure is well understood and that MP&L has taken appropriate corrective action. Therefore, PNL concludes that the low-pressure fuel lines are properly supported and are acceptable for the first refueling cycle.

Component: Crankcase Capscrews

Brief History of Failures

During a 24-hour run of the Division II engine on March 15, 1982, the generator was damaged by the head of a 15/16-inch crankshaft capscrew that broke off, found its way into the generator, and became embedded in the stator.

MP&L analysis of the capscrew concluded that the failure was due to a low-cycle stress fatigue front expanding from an initial small crack. The failed capscrew also had a decarburized skin, which may have contributed to the failure. Vibratory tests indicate that vibrations during startup and shutdown may be contributory to capscrew failure.

MP&L Status

MP&L has installed protective screens at the generators of both Division I and II engines. MP&L has also provided for proper preload of crankcase capscrews to be measured periodically.

PNL Conclusions

PNL concludes that, although capscrews may continue to fail from time to time, this no longer represents a problem for the generators because the protective screen has been installed to prevent broken capscrews from entering the generator. Therefore, PNL recommends that the crankcase capscrews be accepted for the first refueling cycle.

Component: Fuel Oil Leaks; Air Start Valve Failures; Air Start Valve Solenoid Failures; Fuel Oil Injection Pump; and Division I Engine Failures to Start

Brief History of Failures

Failures of all the above five items were recorded in GGNS Division I and II engine logs.

Cause of Failure and Utility Status

All the above items were discussed at the July 13, 1984, meeting among NRC, MP&L, and PNL. For each issue MP&L orally explained the cause of the problem and corrective action taken. MP&L agreed to furnish NRC with documentation on the cause of the failures and the corrective action taken.

PNL Conclusions

PNL considers the information provided orally on July 13, 1984, to be reasonable. That is, MP&L has adequately determined the causes of the problems and has taken appropriate actions to correct them. PNL considers the forthcoming MP&L documentation of resolution of the five items to be confirmatory to the July 13 discussions and concludes that these items should not prevent the Division I engine from being accepted for the first refueling cycle.

4.0 ANALYSIS OF THE REQUIREMENT FOR DIVISION II ENGINE INSPECTION

In the Safety Evaluation Report accompanying the NRC Order of May 22, 1984, requiring diesel generator inspection, the NRC staff stated that the need for Division II engine inspection would be contingent upon:

1. results of the inspection of the Division I engine
2. MP&L's ability to demonstrate, through a review of the manufacturer's QA records, that the two engines have similar "as-manufactured" quality.

4.1 DIVISION I ENGINE INSPECTION RESULTS

Conclusions reached by PNL regarding the Division I engine inspection are provided in Section 3 of this TER. In summary, the Division I engine can reliably serve as a standby power source for the first refueling cycle, subject to load limitations and supported by an enhanced surveillance and maintenance program.

4.1.1 PNL Evaluation

The PNL onsite inspection and the MP&L report of July 5, 1984, revealed only one component, the turbocharger, in which failed elements, bolts and a vane, might be expected to occur in the Division II engine. The other components showed no rejectable indications or incipient problems that suggested adverse conditions might be present in the Division II engine.

4.1.2 PNL Conclusion

The turbochargers from the Division II engine should be inspected, any corrective actions taken, and findings documented. No other Division II inspections are recommended on the basis of the Division I results.

4.2 ENGINE SIMILARITY DEMONSTRATION

MP&L performed a review and assessment that included the following considerations:

- the similarity of the design and as-manufactured quality of the two diesel engines
- the similarity of the post-manufactured upgrades accomplished for each of the two engines
- a comparison of the operating history and operational performance of the two engines
- a comparison of the results of the previous inspections of the two engines.

4.2.1 PNL Evaluation

The "comparability" review was thorough and did not reveal any engine components where differences between Division I and II would significantly affect the Division II engine performance. It was reported that the crankshafts were manufactured by different vendors. Both vendors are judged adequate by the PNL consultants. The difference noted in the oil hole fillets (7/16 inch in Division I versus 3/16 inch in Division II) was noted. MP&L stated in the July 5, 1984, submittal that FaAA analysis concluded that oil hole radius contributes little to the stress concentration. The PNL consultants believe this conclusion is reasonable.

The engine upgrades (installation of AE piston skirts and friction-welded push rods) on Division I were also implemented on Division II. Thus, the two engines are comparably equipped.

The engine operating records supplied by MP&L in the July 5, 1984, submittal indicate that the Division II engine has about 66% fewer starts and 36% less run time than Division I. Further, there is no pattern to valid failures to start that would suggest the Division II engine is significantly less reliable than Division I. PNL notes, however, that the connecting rods have been subjected to approximately 200 hours of operation since the bolt preloading was last checked.

4.2.2 PNL Conclusions

On the basis of the review conducted by MP&L on the manufacturer's QA records and the upgrade accomplished for both engines, PNL concludes that the

Division I and II engine components are of comparable "as-manufactured" quality. On the basis of the operating history, PNL concludes that the engines have been assembled and maintained comparably and the Division II engine has seen less service. Based on these factors and the absence of adverse findings from the recent inspection of the Division I engine, the Division II inspections can be limited to verifying the Division II connecting rod bolt preloading and inspecting the Division II turbocharger, as identified in Section 4.1.2 above.

PNL assumes that MP&L will implement the same enhanced surveillance and maintenance program on the Division I and II engines to maintain their equivalence.

5.0 REVIEW OF THE POST-INSPECTION TESTING

The NRC Order of May 22, 1984, required post-inspection testing to confirm the engines' operability. The testing requirements included the engine manufacturer's recommended preoperational test and additional tests as follows:

- 10 modified starts[a] to 40% load (i.e., 40% of nameplate rating)
- 2 fast starts[b] to 70% of nameplate rating
- one 24-hour run at 70% of nameplate rating.

MP&L's letter (AECM-84/0325) to NRC of July 2, 1984, provided NRC with MP&L clarifications/interpretations of the required testing. The tests accomplished are:

- 10 modified starts to 50% load
- 2 fast starts, started manually from the control room with demonstrated load sequencing and shedding, to 70% load
- one 24-hour run at 70% load.

5.1 PNL EVALUATION

MP&L reported successfully accomplishing all engine manufacturer-recommended post-maintenance testing and all NRC required testing. PNL had understood the fast starts would be done without manual prelubing of the turbochargers. However, the MP&L clarification/interpretation letter (AECM-84/0325) dated July 2, 1984, stated that "all engine starts required by the Order will be preceded by a prelube period...". Such starts are not recognized as simulating starts accompanying loss of offsite power.

5.2 PNL CONCLUSIONS

PNL concludes that post-inspection testing was satisfactorily accomplished with the exception that the fast starts did not simulate the worst challenge to the turbocharger bearings. PNL does not recommend additional testing to

---

(a) A modified start is a start including turbocharger prelube and a 3- to 5-minute loading to the specified load and run for a minimum of one hour.
(b) A fast start simulates ESF signal with the engine in ready-standby status.

simulate this challenge. The information cited earlier in this report for turbocharger thrust bearings provides assurance that the number of "dry" starts anticipated by MP&L is small (two per year per engine), and that the thrust bearings may reasonably be expected to operate satisfactorily for many more than the anticipated number of "dry" starts through the first refueling cycle.

6.0 REVIEW OF THE PROPOSED AUGMENTED MAINTENANCE/SURVEILLANCE PROGRAM

In a letter dated April 16, 1984, to C. Berlinger, PNL identified elements of a maintenance/surveillance (M/S) program that would provide added assurance that the performance of key components of the GGNS TDI engines would be regularly reviewed and that early data would be available to detect potential component failures. It was felt that, in the absence of the completed Owners' Group Program Plan, enhanced M/S is needed to ensure engine reliability. Clarification of some elements of the M/S program was provided to NRC in a letter to C. Berlinger dated April 17, 1984. Subsequently, the features of the enhanced M/S program suggested by PNL were incorporated by the NRC staff in a letter to MP&L dated April 25, 1984.

The MP&L submittal of July 5, 1984, proposed an augmented M/S program for the GGNS Unit 1 diesel engines. MP&L proposed that this revised program remain in effect "...until such time that the reliability of the TDI engines has been demonstrated as adequate by MP&L and the TDI D/G Owners' Group to the satisfaction of the NRC." The MP&L proposed program differs somewhat from the NRC staff recommendations. The differences are aimed at reducing the time that the engines would not be available while the GGNS is at power. Table 1 provides a comparison of the NRC and MP&L M/S program elements.

6.1 PNL EVALUATION

PNL has recommended that utilities seeking licensing prior to the Owners' Group completing all elements of their plan should provide for enhanced surveillance and maintenance (see Section 4 of PNL-5161). Generally, MP&L has provided this. However, as evidenced in Table 1, there are significant differences between the NRC guidance of April 25 and the July 5 proposal by MP&L.

TABLE 1. Comparison of NRC and MP&L Proposed Maintenance/Surveillance for Key Components of the GGNS TDI Engines

Component	NRC Guidance (April 25)	MP&L Proposal (July 5)
Cylinder heads	Air roll 4 hours after engine runs and each day thereafter	Air roll 4 hours after engine runs and each week thereafter
Engine block and base	Visually inspect after 24 hours operation or monthly	Same as NRC
Connecting rods	Visually inspect and retorque after 24 starts, 50 hours operation, or 6 months, whichever is first	Visually inspect and retorque after 50 starts, 270 hours operation, or at the first refueling outage, whichever is first
Lube oil check	Check for water following preoperational tests, then weekly or after 24 hours operation, whichever is first. Check monthly for contaminants and water in sump; check filters	Monthly checks
Studs/fixtures	Check 25% monthly for torque	Check 25% after 270 hours or at the first refueling outage, whichever is first
Push rods, cams, tappets, etc.	Visually inspect after 24 hours operation	Visually inspect after 270 hours operation or at the first refueling outage, whichever is first
Other M/S items	<u>Standby:</u> Lube oil filter differential pressure - daily	<u>Standby:</u> Lube oil filter differential pressure - hourly
	Crankshaft deflections - 6 months	Crankshaft deflection - after 270 hours or at refueling
	<u>Operations:</u> Exhaust temp. - continuous (record hourly)	<u>Operations:</u> Generally per NRC guidance
	Lube oil, jacket water, interlock temp., air pressure, accelerometers - continuous (record hourly)	

6.1.1 Cylinder Heads

The engine air-roll is to detect water in the cylinder, indicating cracked cylinder heads. Water in the cylinder would seriously impact engine operability. The MP&L proposal is to air roll weekly rather than daily to reduce engine unavailability. PNL does not consider this proposal to be adequate for assuring timely detection of water in the cylinders. A revised schedule of air rolls, including one each at 4 and 24 hours after engine shutdown and, thereafter, prior to planned engine starts, is recommended. The basis for the change from the earlier PNL recommendation (which called for rolling the engine every 24 hours) is the recognition that, if a leak has not occurred before 24 hours downtime, it is unlikely that one will be generated before the next time the engine is operated.

6.1.2 Connecting Rods

The visual inspection and retorquing are to provide assurance that the serrated rod joint has not loosened, which could lead to engine failure. The relevant Owners' Group report (FaAA-84-3-14) recommends that the bolt retorquing interval not exceed 200 hours at full load, 248 hours at 85% load, and 286 hours at 75% load. The Owners' Group does not differentiate between conrods having 1-1/2-inch bolts and those having 1-7/8-inch bolts (the latter having higher stresses). However, the GGNS conrods have the 1-7/8-inch bolts; theirs is the only V-16 engine in nuclear service with bolts of this size. Add to these factors the observation of some minor fretting in the serrated joints, noted in connection with the latest engine inspection, and a retorquing approach more conservative than that proposed by MP&L is recommended. A retorquing schedule of 200 hours of operation or 9 months, whichever occurs first, is considered adequate. The 200-hour retorquing interval (rather than the earlier proposed 50-hour interval) is based on PNL's review of the Owners' Group report and the MP&L analysis of the adverse impact of more frequent inspections on engine availability.

6.1.3 Lube Oil Checks

Lube oil checks serve two main functions: they indicate water in the oil that can lead to early engine failures (as well as indicating cracks in engine components), and they may be useful for detecting abnormal wear of engine parts. In this last regard it is important to collect the lube oil sample while the engine is running; MP&L did not specifically provide for this. Otherwise, the proposed monthly rather than weekly lube oil check is considered sufficient, in light of reevaluation based on the experience of the PNL diesel engine consultants.

6.1.4 Studs/Fixtures

Loss of preload on studs can affect engine operability if it goes unnoticed. The air start valve capscrews are more susceptible to loss of preload than are the other threaded fasteners because the gasket material used with these capscrews is softer. One consequence of loss of preload may be loss of cylinder compression.

The MP&L proposed schedule of retorquing on a 25% sampling basis at 270 hours or at the first refueling outage is considered acceptable, based on the judgment of the PNL diesel engine consultants, with the exception of the air start valve capscrews. All (100%) of these capscrews should be retorqued on the MP&L frequency.

6.1.5 Push Rods, Cams, Etc.

Engine operability is affected by defects in push rods, cams, and other similar components. Periodic visual inspection is therefore needed. The difference between the NRC guidance (after 24 hours operation) and MP&L proposal (after 270 hours operation or at the first refueling, whichever is first), is not considered significant in light of the low wear rates of these components, because all parts have been inspected and because, in the opinion of the PNL consultants, there is very little chance of changes in the condition of these parts taking place in the 270-hour (versus the 24-hour) time period. Therefore, the MP&L proposal is considered acceptable.

Additional Surveillance

Surveillance of a number of key engine parameters is essential to assuring reliable engine performance. The NRC guidance and MP&L proposed surveillance are generally quite similar. The differences noted in frequency of measuring lube oil pressure difference and hot and cold crankshaft deflection are not of major significance; thus, the MP&L proposals in these areas are acceptable.

Some clarification of the terms used in the MP&L July 5, 1984, submittal is recommended. Also, one item of surveillance, engine load, was not addressed. The following changes in Section 6.7 of the MP&L submittal are therefore recommended:

p. 57, Discussion - add the word "hourly" after "recorded" in line 2.

p. 58 - replace as noted:

- "lube oil pressure" to "engine inlet lube oil pressure"
- "combustion air L.B. pressure" and "combustion air R.B. pressure" to "air manifold pressure L.B. and R.B."
- "jacket water pressure" to "jacket water pressure in and out"
- "cylinder temperatures" to "all cylinder exhaust temperatures"
- "stack temperatures" to "preturbine exhaust temperatures"
- add "engine load" as a new item.

p. 59, MP&L Proposed Action - add "or each refueling cycle, whichever occurs first," after "operation" in line 3.

p. 59 - Add a new item of surveillance, namely "check the rotor float of at least one turbocharger and inspect stationary nozzle ring bolts, after 270 hours of operation or at the first refueling outage, whichever comes first."

p. 64, Table 6-2 - add "clear water system (flush out)" with frequency of 3 to 4 years.

6.2 PNL CONCLUSIONS

PNL concludes that the MP&L proposed M/S activities need some modifications to provide adequate assurance of engine reliability/operability. The modifications are discussed in detail above in Section 6.1. In summary they are:

- cylinder heads - Revise air roll to 4 and 24 hours after each engine shutdown and prior to planned engine starts.

- connecting rods - Revise retorquing frequency to 200 hours or 9 months, whichever occurs first. A retorquing check should be performed on the Division II engine prior to plant operation.
- lube oil checks - Add that a lube oil sample will be obtained while engine is running.
- studs/fixtures - Modify to assure that 100% of the air start valve capscrews will be retorqued on the schedule indicated.
- additional surveillance - Provide changes as detailed above in Section 6.1.6.

With these modifications, the MP&L proposed M/S activities are considered acceptable for the first refueling cycle.

7.0 OVERALL CONCLUSIONS

PNL and its consultants conclude that the TDI diesel engines at the GGNS have the needed operability and reliability to fulfill their intended (auxiliary) emergency power function for the first refueling cycle. This conclusion is reached with a number of understandings regarding 1) limits to the engine requirements, 2) NRC concurrence with MP&L findings/conclusions regarding items to be supplied to NRC, 3) limitations on the engine Brake Mean Effective Pressure (BMEP), and 4) MP&L's implementation of the modifications to their proposed surveillance and maintenance program identified in Section 6. Further details on these items follow.

7.1 LIMITED ENGINE REQUIREMENTS

PNL understands that the emergency service requirements MP&L now foresees for the GGNS will not exceed the engine load corresponding to a BMEP of 185 psig.

7.2 NRC CONCURRENCE WITH ADDITIONAL MP&L SUBMITTALS

The PNL conclusion that the TDI engines will provide adequate standby power for the GGNS is predicated on an understanding that a technical review of the following MP&L submittals to NRC will not raise unanticipated problems:

- an inspection report confirming that the turbocharger turbine nozzle bolt failure was due to intergranular stress corrosion
- a submittal describing in detail the method used and the results to confirm the surface area contact of the serrated surfaces of each connecting rod is at least 75%
- documented results of measurements of the cylinder head firedeck surface flatness
- the inspection and engineering evaluation reports confirming the acceptability for continued service of the two cylinder heads that contain cracks in the stellite seats
- a submittal identifying the design of cylinder head replacement studs

- MP&L documentation of the indications noted and the engineering disposition concerning the relative motion between the piston crown and skirt
- documented crankshaft deflections relative to TDI specifications
- crankshaft torsiographs at 0%, 25%, 50%, 75% and 100% of engine nameplate loading and associated stresses as identified in a PNL letter to NRC dated July 17, 1984
- documented preturbine exhaust temperatures relative to the manufacturer's recommended maximum.

7.3 ENGINE BMEP LIMITATIONS

PNL understands that all subsequent engine testing (except the abovementioned torsiograph at 100% loading and the test to obtain preturbine exhaust temperature data) will be limited to the load corresponding to 185 BMEP.

7.4 REVISED SURVEILLANCE/MAINTENANCE PROGRAM

PNL understands that MP&L will resubmit to NRC a revised surveillance and maintenance plan incorporating the recommended changes identified in Section 6 of this report.

NRC FORM 335 (7-77) U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET**	1. REPORT NUMBER (Assigned by DDC) NUREG-0831 Supplement No. 6
4. TITLE AND SUBTITLE (Add Volume No., if appropriate) Safety Evaluation Report related to the operation of Grand Gulf Nuclear Station, Units 1 and 2	2. (Leave blank)
	3. RECIPIENT'S ACCESSION NO.
7. AUTHOR(S)	5. DATE REPORT COMPLETED MONTH August YEAR 1984
9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) Division of Licensing Office of Nuclear Reactor Regulation U.S. Nuclear Regulatory Commission Washington, D. C. 20555	DATE REPORT ISSUED MONTH August YEAR 1984
	6. (Leave blank)
	8. (Leave blank)
12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code) Same as 9 above	10. PROJECT/TASK/WORK UNIT NO.
	11. CONTRACT NO.
13. TYPE OF REPORT Safety Evaluation Report Supplement 6	PERIOD COVERED (Inclusive dates)
15. SUPPLEMENTARY NOTES Pertains to Docket Nos. 50-416 and 50-417	14. (Leave blank)

16. ABSTRACT (200 words or less)

Supplement No. 6 to the Safety Evaluation Report for Mississippi Power & Light Company et al joint application for licenses to operate the Grand Gulf Nuclear Station, Units 1 and 2, located on the east bank of the Mississippi River near Port Gibson in Claiborne County, Mississippi, has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement reports the NRC staff's evaluation of open items from previous supplements and Technical Specification changes required before authorizing operation of Unit 1 above 5% of rated power.

17. KEY WORDS AND DOCUMENT ANALYSIS	17a. DESCRIPTORS
17b. IDENTIFIERS/OPEN-ENDED TERMS	

18. AVAILABILITY STATEMENT	19. SECURITY CLASS (This report)	21. NO. OF PAGES
Unlimited	Unclassified	
	20. SECURITY CLASS (This page) Unclassified	22. PRICE $

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

FIRST CLASS MAIL
POSTAGE & FEES PAID
USNRC
WASH D C
PERMIT No G67

120555078877 1 1AN
US NRC
ADM-DIV OF TIDC
POLICY & PUB MGT BR-PDR NUREG
W-501
WASHINGTON DC 20555